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ORNL-3820
UC-41 – Health and Safety
TID-4500 (41st ed.)

APPLIED HEALTH PHYSICS

ANNUAL REPORT FOR 1964

p10; 28 Ci I-131 in a wk released

OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

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Printed in USA. Price $3.00. Available from the Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U.S. Department of Commerce, Springfield, Virginia

ORNL-3820

Contract No. W-7405-eng-26

HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS ANNUAL REPORT FOR 1964

K. Z. Morgan, Director

D. M. Davis, Section Chief

JUNE 1965

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION

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The data for this report were contributed by: H. H. Abee, Section Chief of the Environmental Monitoring Section; R. L. Clark, Section Chief of the Radiation Survey Section; E. D. Gupton, Section Chief of the Dosimetry and Meters Section; and A. D. Warden, Associate Department Head of the Applied Health Physics Department.

Grateful acknowledgment is made to J. C. Hart, Consultant, for assisting in editing this report.

4.0 SUMMARY

The waste gaseous and liquid releases from the Laboratory were such that the concentration of radioactive materials in the environs were well below the maximum levels recommended by the NCRP and FRC. The average concentration of radioactive materials in the atmosphere at the perimeter of the controlled area was less than one percent of the maximum permissible. The number of beta curies released via White Oak Creek to the Clinch River was 234, or about half the 1963 release of 469 curies. The average concentration of ^{131}I in raw milk samples was 11.4 pc/l or about 12 percent less than the 1963 value. Background measurements at five selected locations adjacent to the ORNL area decreased by a factor of about two from 1963 to 1964. The average for 1964 was 0.014 mR/hr.

No employee received an external or internal radiation dose which exceeded the maximum permissible levels recommended by the FRC. The highest whole body dose equivalent received by an employee was about 4.2 rem or 35 percent of the maximum permissible annual dose. During 1964 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than half of a maximum permissible body burden.

During 1964 the frequency of Unusual Occurrences dropped to a five-year low of 29 events. Only about 50 percent of these events were classified as significant according to the ORNL rating of Unusual Occurrences.

Revisions in personnel monitoring techniques and development of improved bio-assay procedures have resulted in significant cost reduction in these programs during 1964.

During 1964, 22 persons from outside the continental United States received training at Applied Health Physics (AHP) facilities. In addition, 130 other persons from within the United States visited AHP for training purposes or to observe AHP activities.

5.0 ENVIRONS MONITORING

The Health Physics Division monitors for airborne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of twenty-two stations which are positioned in relation to ORNL operational activities (Figures 1 and 2); the perimeter air monitoring (PAM) network consists of nine stations which are located on the perimeter of the AEC controlled area (Figure 3); and the remote air monitoring (RAM) network consists of seven stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL (Figure 4). The monitoring networks provide for the collection of (1) airborne radioactivity by air filtration techniques, (2) radioparticulate fallout material by impingement on gummed paper trays, and (3) rain water for measurement of fallout occurring as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fallout; and the rain water data provide information on the soluble and insoluble fractions of the radioactive content of fallout material.[1]

Low level radioactive liquid wastes originating from ORNL operations are discharged, after preliminary treatment, to White Oak Creek, which is a small tributary of the Clinch River. Liquid waste releases are controlled so that the resulting average radioactive concentrations in the Clinch River comply with maximum permissible concentrations established for populations in the neighborhood of an atomic energy installation as recommended by the National Committee on Radiation Protection (NCRP) and the Federal Radiation Council (FRC).

The radioactive content of the White Oak Creek discharge is determined at White Oak Dam (Figure 5) which is the last control point along the stream prior to entry of White Oak Creek waters into Clinch River waters. Water samples are collected at a number of locations along the Clinch River, beginning at a point above the entry of wastes into the river via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the confluence of White Oak Creek and the Clinch River. Water samples are analyzed for gross radioactivity and for certain specified long-lived radionuclides. From the maximum permissible concentration values for drinking water, $(MPC)_w$, for each isotope as recommended by NCRP, a weighted average $(MPC)_w$ for the mixture of radionuclides is calculated on the basis of the isotopic distribution in the water. The average concentrations of gross activity are used for control purposes.

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[1]A detailed discussion concerning techniques used in processing air and water samples for environmental monitoring purposes is given in ORNL-2601, "Radioactive Waste Management at Oak Ridge National Laboratory".

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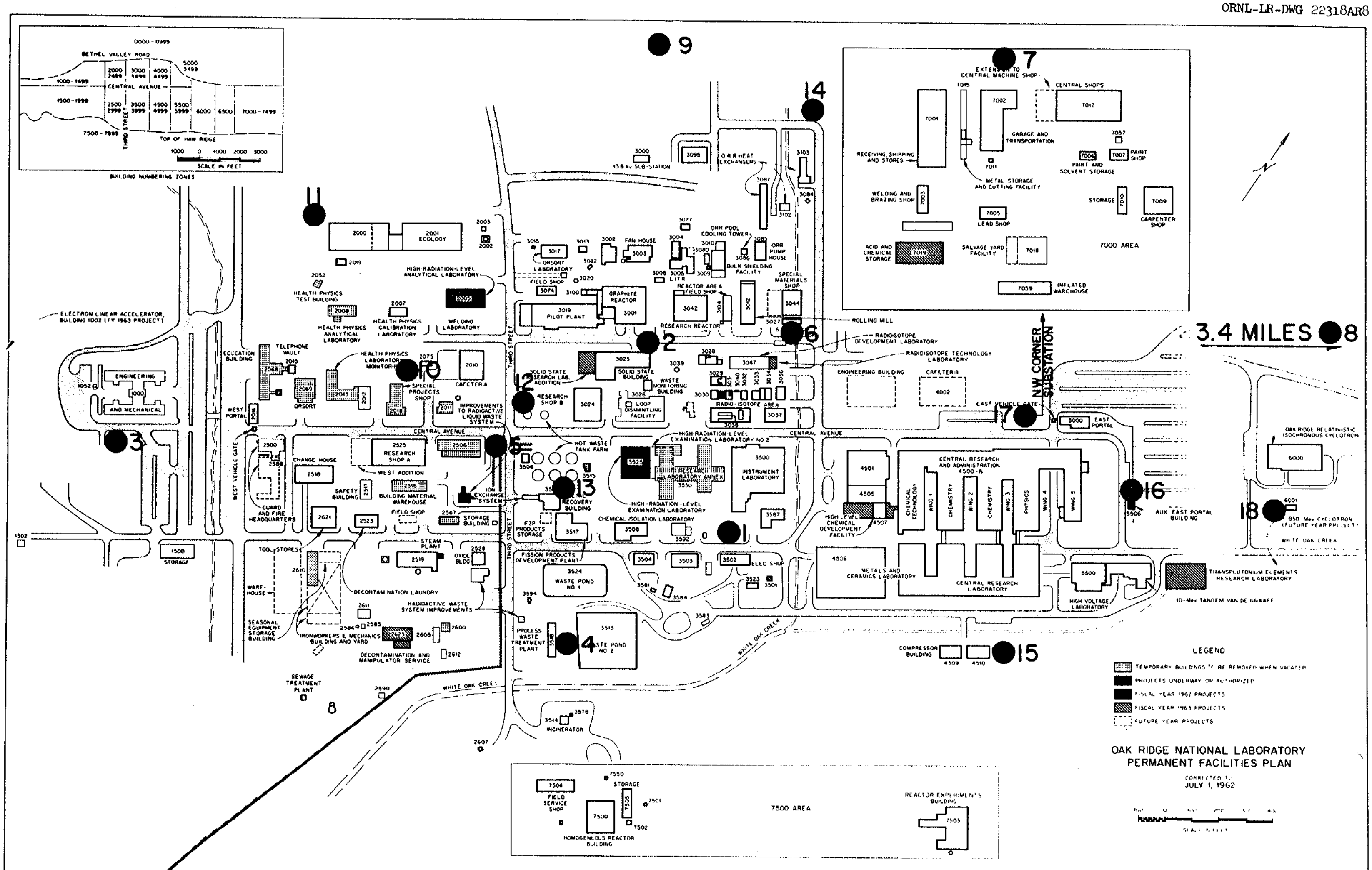

Fig. 1 Map of X-10 Area Showing the Approximate Location of 18 of 22 of the Local Monitoring Stations Constituting the LAM Network.

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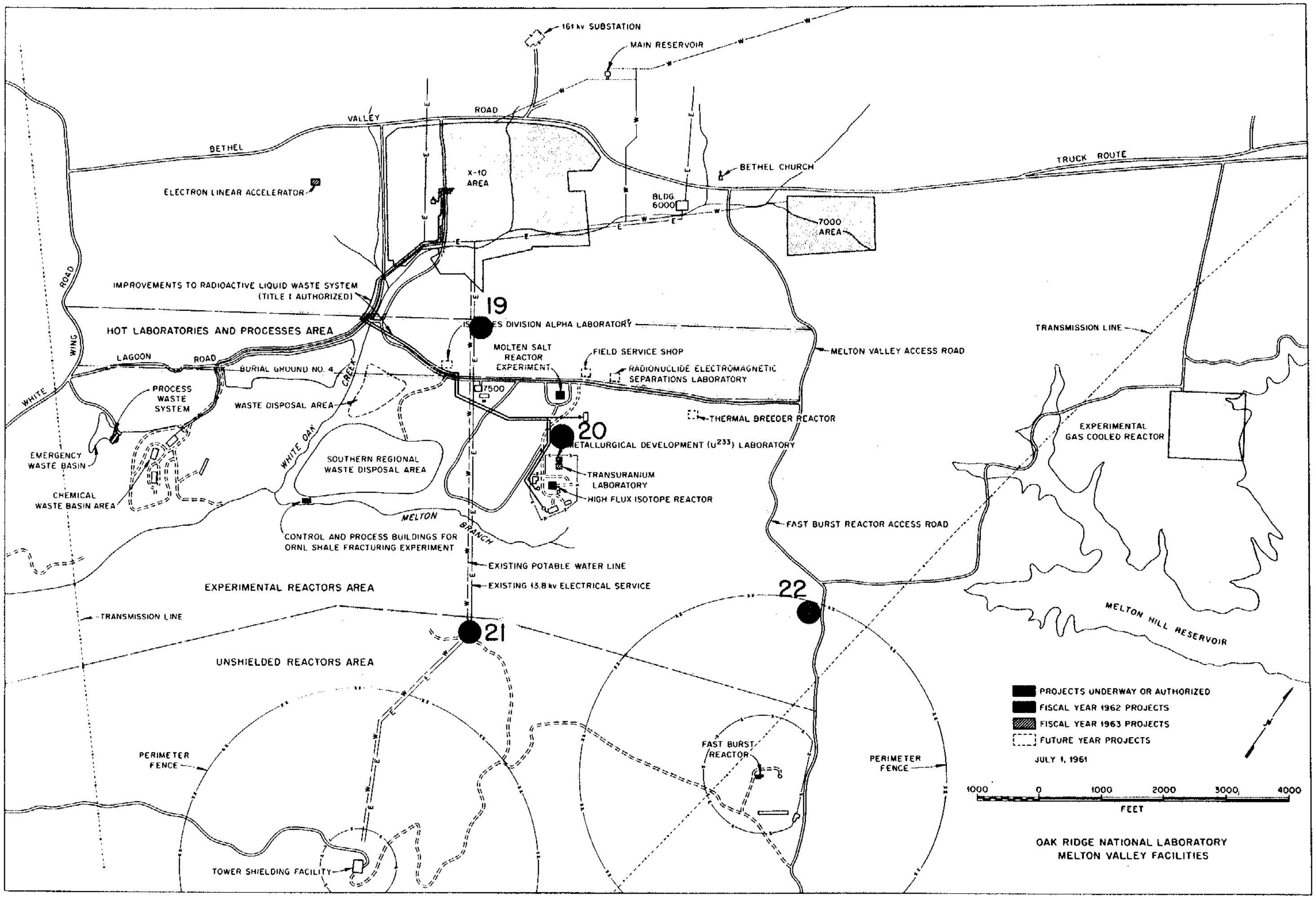

Fig. 2 Map of Laboratory Area Showing the Approximate Location of 4 of the 22 Local Monitoring Stations Constituting the LAM Network.

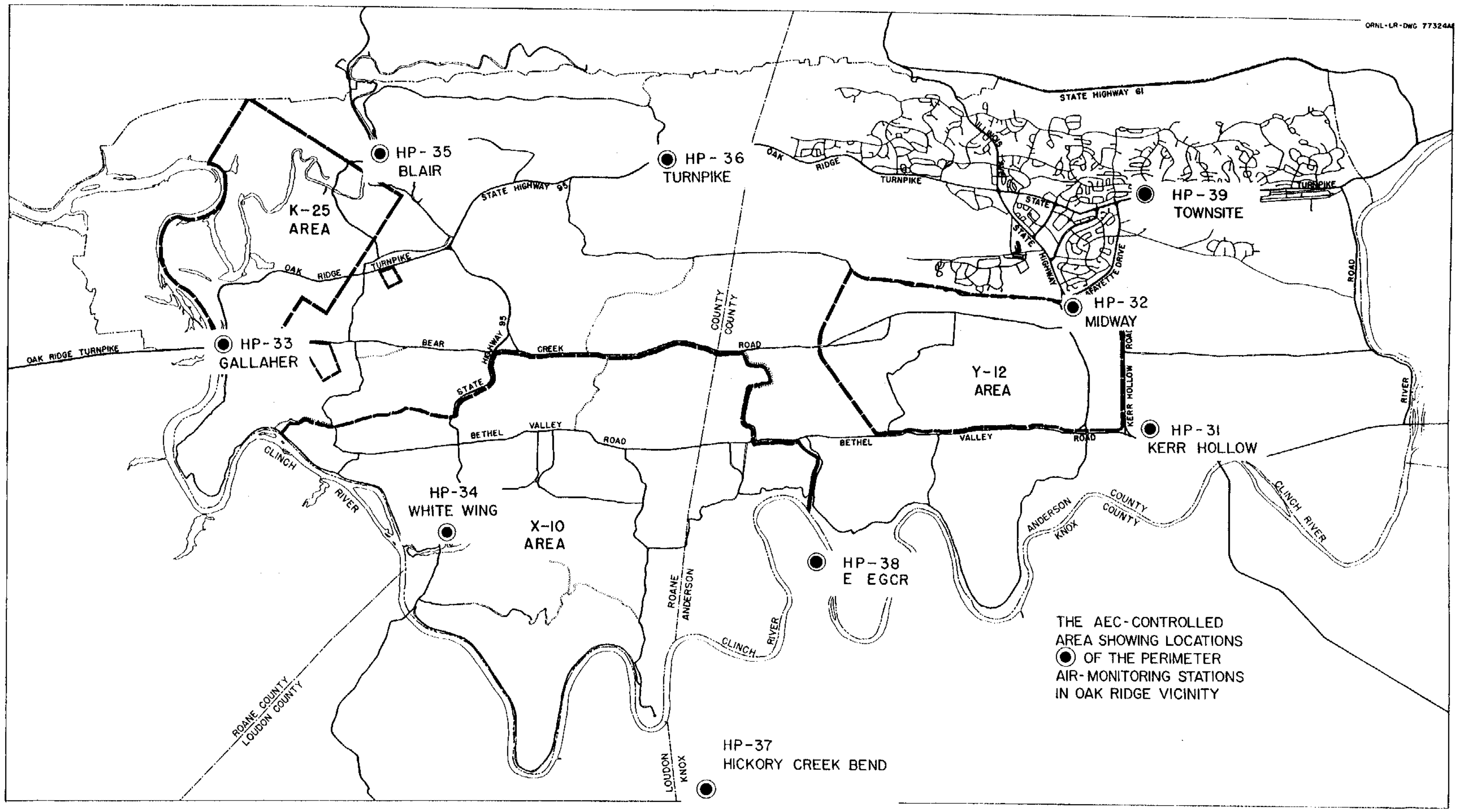

Fig. 3 Map of the AEC Controlled Area and Vicinity Showing the Approximate Location of the Perimeter Air Monitoring Stations Constituting the PAM Network.

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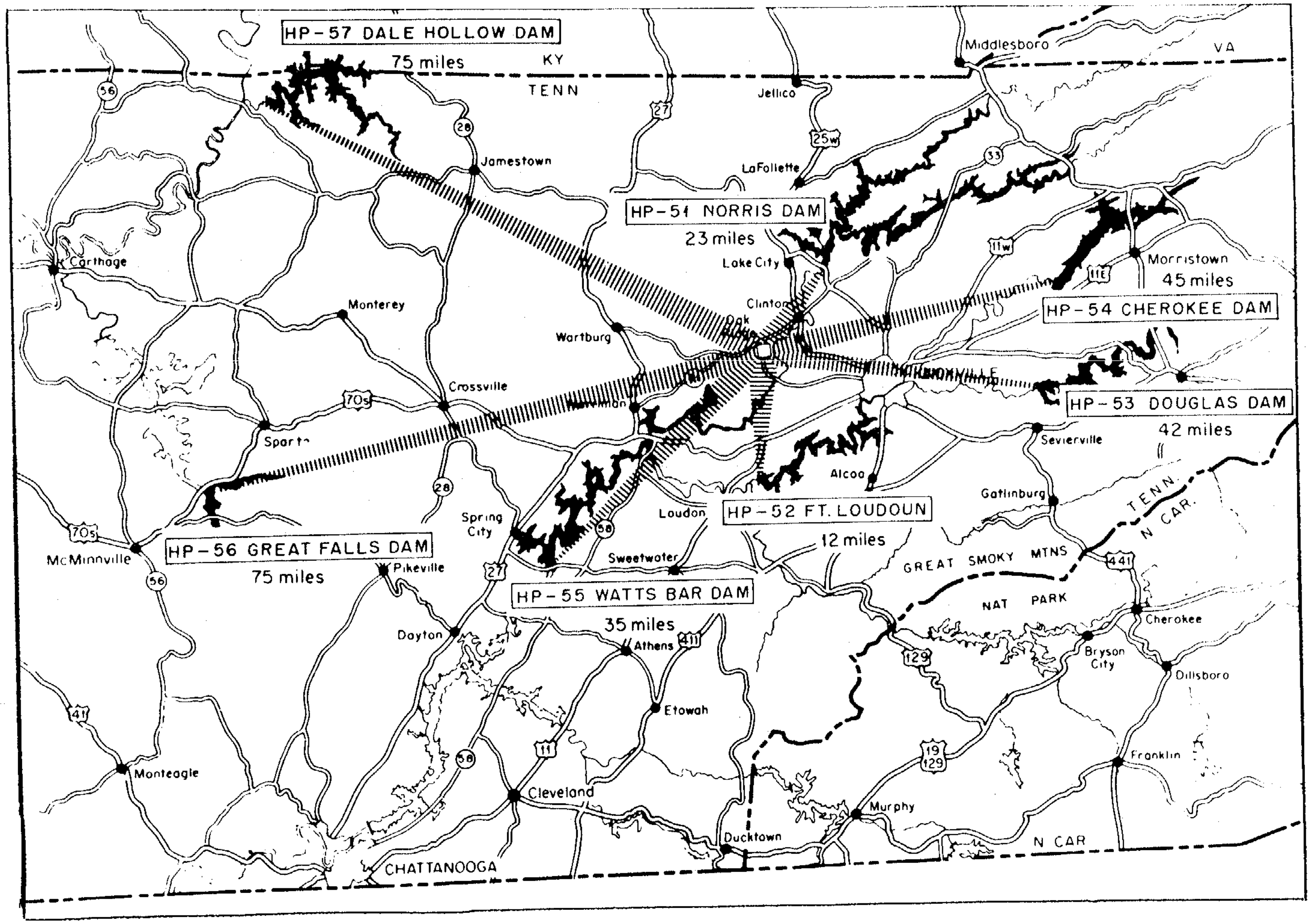

Fig. 4 Map of a Section of the East Tennessee Area Showing TVA and U.S. Corps of Eng. Dam Sites at which are Located the Remote Air Monitoring Stations Constituting the RAM Network.

ORNL-DWG. 64-698

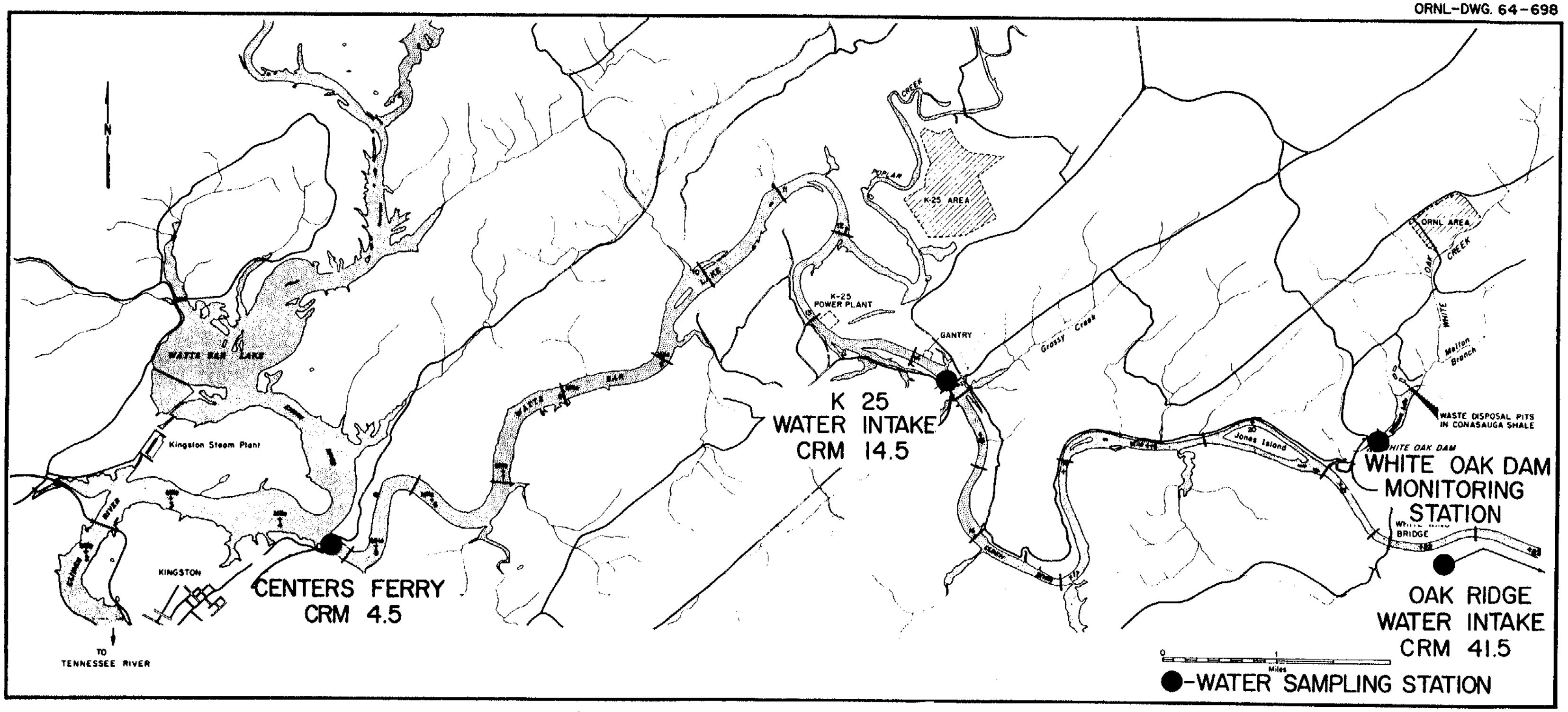

Fig. 5 Map Showing Water Sampling Locations in the East Tennessee Area.

Samples of ORNL potable water are collected, composited and stored at the rate of one sample per day. At the end of each quarter these composites are analyzed radiochemically for ^{90}Sr content and are assayed for long-lived gamma emitting radionuclides by gamma spectrometry.

Raw milk samples are collected at twelve sampling stations located within a radius of 50 miles from ORNL. Samples are taken on a weekly basis from eight stations which are located outside the AEC controlled area within a 12-mile radius of ORNL (Figure 6). Samples are collected every five weeks from the four remaining stations, all of which are located outside the 12-mile radius up to distances of about 50 miles. The purpose of the milk sampling program is twofold: first, samples collected in the immediate vicinity of ORNL provide data by which one may evaluate the possible effect of waste releases originating from ORNL operations; second, samples collected remote to the immediate vicinity of the ORNL area provide background data which are essential in establishing a proper index from which the intentional or accidental release of radioactive materials originating from Oak Ridge operations may be evaluated.

Aerial background surveys are made at least once each calendar quarter over the ORNL area and for several miles distant from ORNL in the general direction of low altitude prevailing winds. Experiments in light aircraft at speeds of approximately 120 miles per hour have shown that, at an altitude of approximately 300 feet, it is possible to detect ^{131}I contamination upon grasslands with reasonable accuracy by scintillation detectors down to levels of about 0.5 $\mu c/m^2$. Thus, light aircraft, equipped with portable scintillation detectors and used in the manner described above, provide a practical means of detecting extensive high-level deposits of ^{131}I on ground surfaces.

Background gamma radiation measurements are made monthly at a number of locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

River bottom sediments in the Clinch and Tennessee Rivers have been surveyed and analyzed annually since the year 1951 for the purpose of providing data relative to the dispersion of radioactive wastes released from Oak Ridge operations to the Clinch River.

5.1 Atmospheric Monitoring

Fallout monitors were installed and activated at the 22 LAM stations. These monitors provide telemetered information to environmental monitoring headquarters regarding levels of fallout under routine and emergency conditions.

Two new perimeter air monitoring stations were added to the network in 1964. Station No. 38 was installed across Clinch River from the EGCR to provide coverage when the EGCR becomes operative. Station No. 39 was located in the city of Oak Ridge.

ORNL—DWG. 64-3713 R1

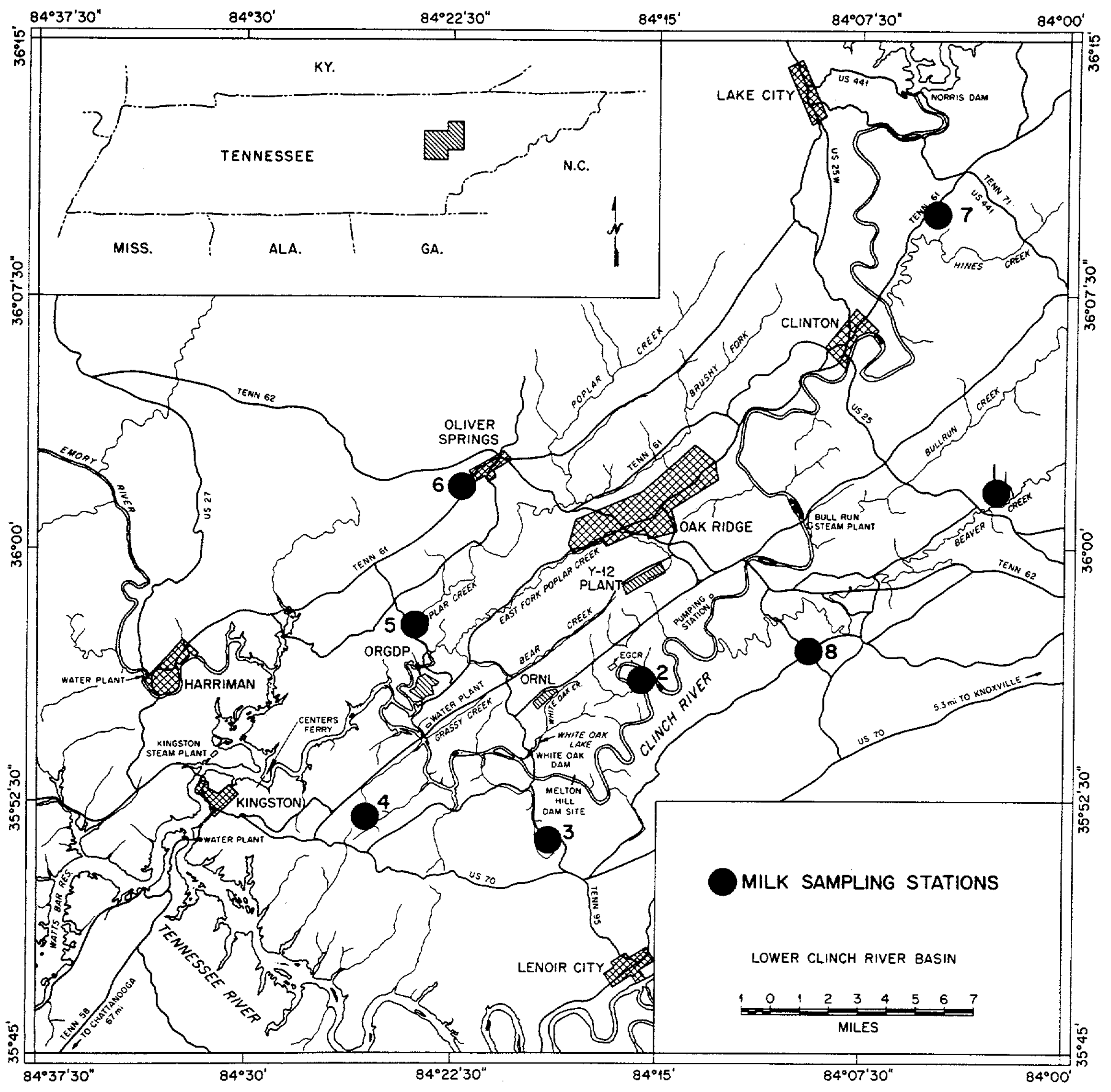

Fig. 6 Map Showing Milk Sampling Stations in the East Tennessee Area.

5.1.1 Air Concentrations - The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM networks during 1964, were as follows:

Network	Concentration (μc/cc)
LAM	1.3×10^{-12}
PAM	0.82×10^{-12}
RAM	1.1×10^{-12}

The LAM network value of 1.3×10^{-12} μc/cc is about 0.13 percent of the $(MPCU)_a$[2] based on occupational exposure. The PAM and RAM network values represent 0.82 and 1.1 percent, respectively, of the $(MPCU)_a$ for persons residing in the neighborhood of an atomic energy installation. A tabulation of data for each station in each network is given in Table 1. The weekly values for each network are graphically illustrated in Figure 7.

5.1.2 Fallout (Gummed Paper Technique) - Radioparticulate fallout as measured by the LAM network of stations decreased by a factor of about 20 from the value measured in 1963. The values measured by both the PAM and RAM networks decreased by a factor of approximately 100 from the 1963 values. LAM network measurements are influenced by fallout from local operations as well as from weapons testing operations and are normally higher than those recorded by the PAM and RAM networks when world wide fallout is low. Table 2 gives a tabulation of data for each station within each network. The weekly average values for each network for each week are illustrated in Figure 8.

The number of radioactive particles collected on air monitor filters of the LAM network in 1964 decreased by a factor of 10 from the number collected in 1963. The values measured by the PAM and RAM networks decreased by factors of 60 and 80, respectively. The average number of radioparticulates per 1000 cubic feet of air sampled at each station in each network is given in Table 1.

5.1.3 Atmospheric Radioiodine (Charcoal Collector Techniques) - Atmospheric radioiodine measured by the perimeter stations averaged 0.03×10^{-12} μc/cc during 1964. This is approximately 0.03 percent of the maximum permissible concentration for populations in the neighborhood of a controlled area. The maximum value observed at any one station for one week was 0.85×10^{-12} μc/cc. This value was measured at the White Oak Dam station and was associated with the release of about 28 curies of radioiodine from ORNL[3] stacks during a period of one week.

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[2] The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. NBS Handbook 69, Table 4, p. 94, gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3] "Summary of Waste Discharges", Week Ending March 22, 1964, L. C. Lasher.

Table 1 CONCENTRATION OF RADIOACTIVE MATERIALS IN AIR - 1964
(Filter Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μc/cc	No. of Particles by Activity Ranges					Particles Per 1000 ft^3
			< 10^5 d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	> 10^7 d/24 hr	Total	
Laboratory Area								
HP-1	S 3587	14	3.2	0.04	0.0	0.00	3.2	0.10
HP-2	NE 3025	18	4.1	0.24	0.0	0.04	4.3	0.15
HP-3	SW 1000	12	1.1	0.00	0.02	0.00	1.1	0.03
HP-4	W Settling Basin	8.8	1.1	0.00	0.00	0.00	1.1	0.04
HP-5	E 2506	17	17	0.27	0.02	0.00	17	0.61
HP-6	SW 3027	11	1.9	0.04	0.00	0.00	2.0	0.09
HP-7	W 7001	14	0.87	0.00	0.00	0.00	0.87	0.03
HP-8	Rock Quarry	13	0.85	0.00	0.00	0.00	0.85	0.02
HP-9	N Bethel Valley Rd.	12	1.4	0.00	0.00	0.00	1.4	0.04
HP-10	W 2075	14	4.3	0.02	0.00	0.00	4.3	0.13
Average		13	3.6	0.06	0.00	0.00	3.6	0.13
Perimeter Area								
HP-31	Kerr Hollow Gate	8.4	1.6	0.00	0.00	0.00	1.6	0.03
HP-32	Midway Gate	10	2.1	0.00	0.00	0.00	2.1	0.04
HP-33	Gallaher Gate	7.2	0.92	0.00	0.00	0.00	0.92	0.02
HP-34	White Wing Gate	7.6	2.0	0.00	0.00	0.00	2.0	0.04
HP-35	Blair Gate	8.9	1.6	0.00	0.00	0.00	1.6	0.03
HP-36	Turnpike Gate	9.2	1.5	0.02	0.02	0.02	1.6	0.03
HP-37	Hickory Creek Bend	8.3	0.85	0.00	0.00	0.00	0.85	0.02
HP-38	E EGCR	3.5	2.3	0.00	0.00	0.00	2.3	0.05
HP-39	Townsite	3.1	0.00	0.00	0.00	0.00	0.00	0.00
Average		8.2	1.5	0.00	0.00	0.00	1.5	0.03
Remote Area								
HP-51	Norris Dam	11	1.3	0.00	0.00	0.00	1.3	0.02
HP-52	Loudoun Dam	11	2.2	0.00	0.00	0.00	2.2	0.04
HP-53	Douglas Dam	11	1.5	0.00	0.00	0.00	1.5	0.02
HP-54	Cherokee Dam	12	3.0	0.00	0.00	0.00	3.0	0.05
HP-55	Watts Bar Dam	10	0.49	0.00	0.00	0.00	0.49	0.01
HP-56	Great Falls Dam	9.2	0.92	0.00	0.00	0.00	0.92	0.02
HP-57	Dale Hollow Dam	10	0.58	0.00	0.00	0.00	0.58	0.01
Average		11	1.4	0.00	0.00	0.00	1.4	0.02

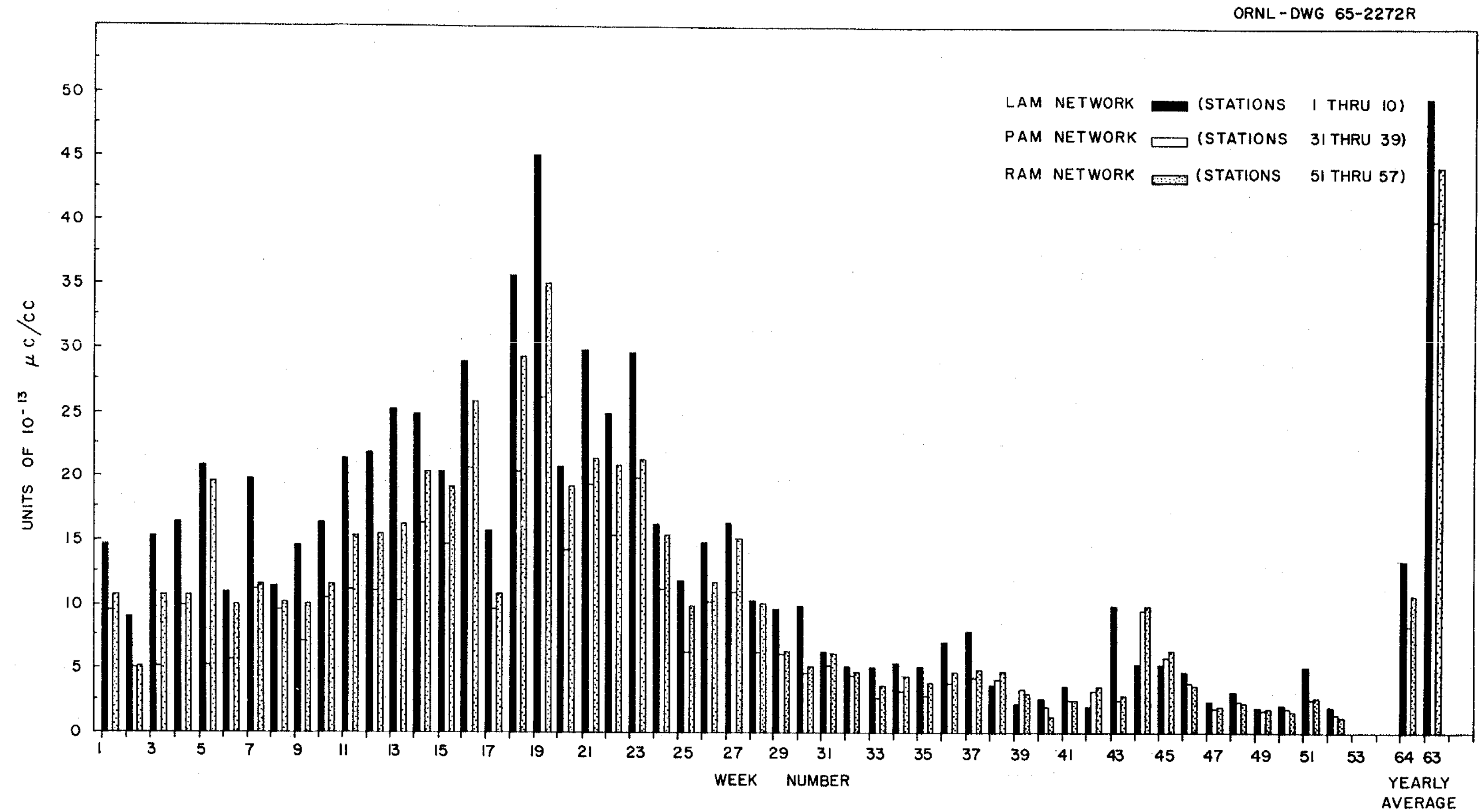

Fig. 7 Concentration of Radioactive Materials in Air as Determined from Filter Paper Data - 1964. (High activity in spring and early summer is attributed to delayed fallout from weapons tests.)

Table 2 RADIOPARTICULATE FALL-OUT - 1964
(Gummed Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-4} μc/ft²	No. of Particles by Activity Ranges $< 10^5$ d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	$> 10^7$ d/24 hr	Total Particles Per Sq. Ft.
			Laboratory Area				
HP-1	S 3587	2.7	0.77	0.04	0.00	0.04	0.85
HP-2	NE 3025	2.5	1.08	0.04	0.02	0.00	1.14
HP-3	SW 1000	2.6	0.21	0.02	0.00	0.00	0.23
HP-4	W Settling Basin	2.4	0.46	0.04	0.00	0.00	0.50
HP-5	E 2506	3.4	2.56	0.19	0.04	0.00	2.79
HP-6	SW 3027	2.3	0.88	0.00	0.00	0.00	0.88
HP-7	W 7001	2.4	0.13	0.00	0.00	0.00	0.13
HP-8	Rock Quarry	2.1	0.02	0.00	0.00	0.00	0.02
HP-9	N Bethel Valley Rd.	2.5	0.25	0.00	0.00	0.00	0.25
HP-10	W 2075	2.3	1.40	0.06	0.00	0.00	1.46
Average		2.5	0.78	0.04	0.01	0.00	0.83
			Perimeter Area				
HP-31	Kerr Hollow Gate	2.6	0.19	0.00	0.00	0.00	0.19
HP-32	Midway Gate	2.6	0.35	0.00	0.00	0.00	0.35
HP-33	Gallaher Gate	2.5	0.10	0.00	0.00	0.00	0.10
HP-34	White Wing Gate	2.0	0.06	0.00	0.00	0.00	0.06
HP-35	Blair Gate	2.6	0.17	0.00	0.00	0.00	0.17
HP-36	Turnpike Gate	2.4	0.10	0.00	0.00	0.00	0.10
HP-37	Hickory Creek Bend	2.3	0.04	0.00	0.00	0.00	0.04
HP-38	E EGCR	0.48	0.00	0.00	0.00	0.00	0.00
HP-39	Townsite	0.44	0.00	0.00	0.00	0.00	0.00
Average		2.27	0.13	0.00	0.00	0.00	0.13
			Remote Area				
HP-51	Norris Dam	2.2	0.13	0.00	0.02	0.00	0.15
HP-52	Loudoun Dam	2.0	0.10	0.00	0.00	0.00	0.10
HP-53	Douglas Dam	1.9	0.02	0.00	0.00	0.00	0.02
HP-54	Cherokee Dam	2.3	0.06	0.00	0.00	0.00	0.06
HP-55	Watts Bar Dam	1.9	0.08	0.00	0.00	0.00	0.08
HP-56	Great Falls Dam	2.2	0.13	0.00	0.00	0.00	0.13
HP-57	Dale Hollow Dam	2.1	0.02	0.00	0.00	0.00	0.02
Average		2.1	0.08	0.00	0.00	0.00	0.08

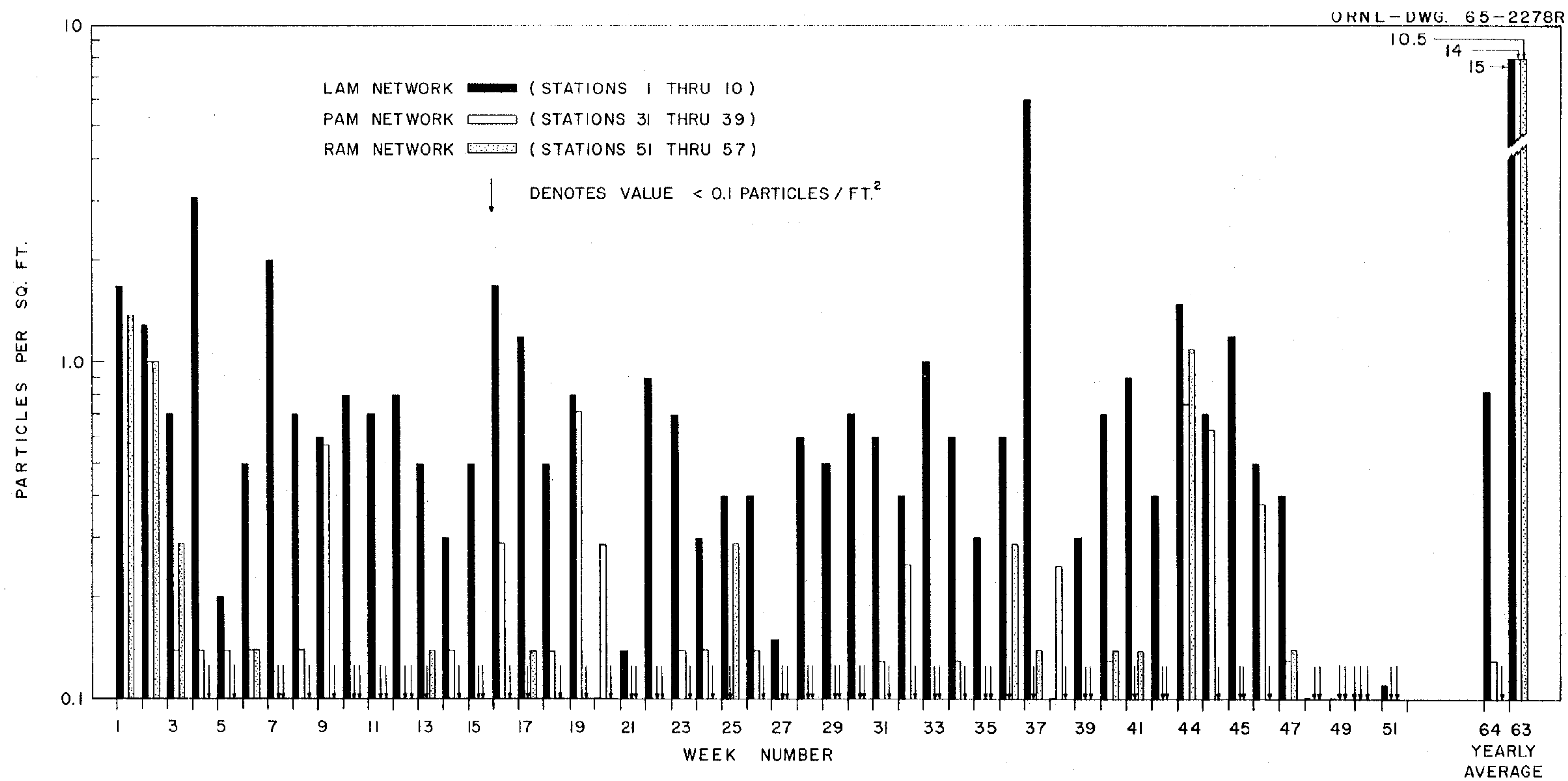

Fig. 8 Radioparticulate Fallout Measurements as Determined by Autoradiographic Techniques Using Gummed Paper Collectors - 1964.

Figure 9 compares the weekly discharge of radioiodine from ORNL stacks[4] with the average concentration of radioiodine measured by the perimeter stations.

5.2 Water Analyses

5.2.1 Rain Water - The average concentration of radioactivity in rain water collected from the three networks during 1964 were as follows:

Network	Concentration ($\mu c/ml$)
LAM	1.5×10^{-7}
PAM	1.5×10^{-7}
RAM	1.7×10^{-7}

These values are lower than those observed during 1963 by a factor of more than 5 on all networks. The lack of a significant difference between network averages indicates that the radioactivity in the rain water was not of local origin. The average values for each station are shown in Table 3; the average values for each network for each week are given in Figure 10.

5.2.2 Clinch River Water - A total of 234 beta curies of radioactivity was released to the Clinch River during 1964 (Table 4). Yearly discharges of radionuclides to Clinch River, 1949 through 1964, are shown in Table 5. Radiochemical analysis of the White Oak Dam effluent indicated that about 82 percent of the radioactivity was ^{106}Ru. The percentage of ^{90}Sr in the effluent was 2.8 compared to 1.7 in 1963.

The calculated average concentration of radioactive materials in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the river) was 1.3×10^{-7} $\mu c/ml$. This represents 2.3 percent of the weighted average $(MPC)_w$ recommended for persons residing in the neighborhood of an atomic energy installation (Table 6). The average concentration of radioactive materials in the Clinch River did not exceed 14 percent of the $(MPC)_w$ during any one week in 1964 (Figure 11).

The measured average concentration of radioactivity in Clinch River water at CRM 41.5 (above the entry of White Oak Creek) was 0.43 percent of the weighted average $(MPC)_w$ (Table 6). The concentration of ^{90}Sr in the river above the entry of White Oak Creek continues to be about the same as the contribution calculated for White Oak Creek effluent at CRM 20.8 assuming uniform mixing of the two streams.

The measured average concentration of radioactive materials in the Clinch River at CRM 4.5 (near Kingston, Tennessee) was 0.57×10^{-7} $\mu c/ml$. This value represents 1.4 percent of the $(MPC)_w$ as applied to persons living in the neighborhood of an atomic energy installation.

[4] "Summary of Waste Discharges", Weekly Reports, 1964, L. C. Lasher.

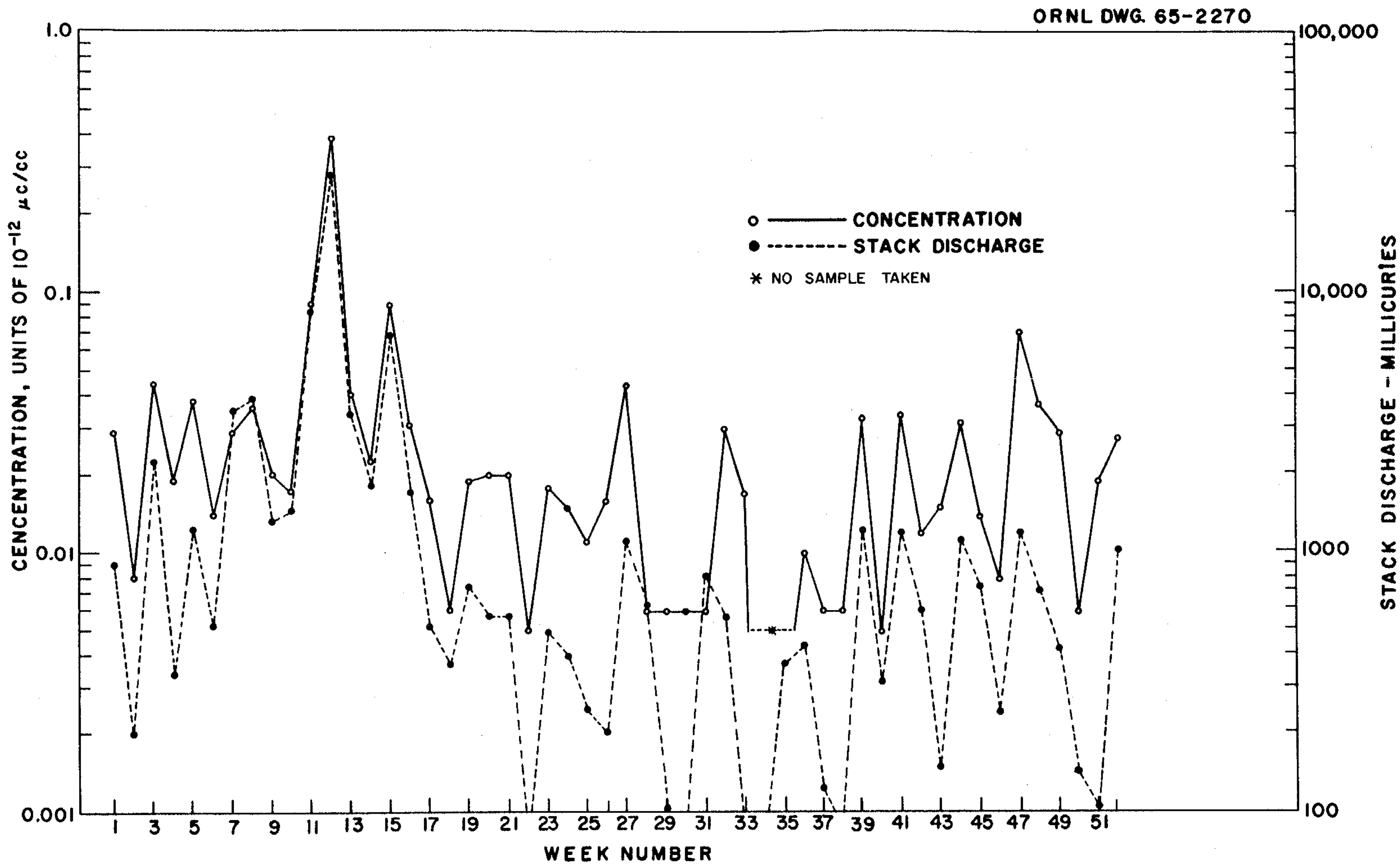

Fig. 9 Weekly Average Concentration of ^{131}I in Air at the Perimeter of the Controlled Area Compared with ^{131}I Discharges from ORNL Stacks - 1964 (Peak during 12th week was due to accidental release of about 20 curies of ^{131}I.)

Table 3 CONCENTRATION OF RADIOACTIVE MATERIALS IN RAIN WATER - 1964
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rain Water, μc/cc
	Laboratory Area	
HP-7	West 7001	1.5×10^{-7}
	Perimeter Area	
HP-31	Kerr Hollow Gate	1.6×10^{-7}
HP-32	Midway Gate	1.5
HP-33	Gallaher Gate	1.5
HP-34	White Wing Gate	1.3
HP-35	Blair Gate	1.8
HP-36	Turnpike Gate	1.6
HP-37	Hickory Creek Bend	1.2
HP-38	E EGCR	0.5
HP-39	Townsite	0.8
Average		1.5×10^{-7}
	Remote Area	
HP-51	Norris Dam	1.9×10^{-7}
HP-52	Loudoun Dam	1.9
HP-53	Douglas Dam	1.7
HP-54	Cherokee Dam	2.1
HP-55	Watts Bar Dam	1.4
HP-56	Great Falls Dam	2.0
HP-57	Dale Hollow Dam	1.3
Average		1.7×10^{-7}

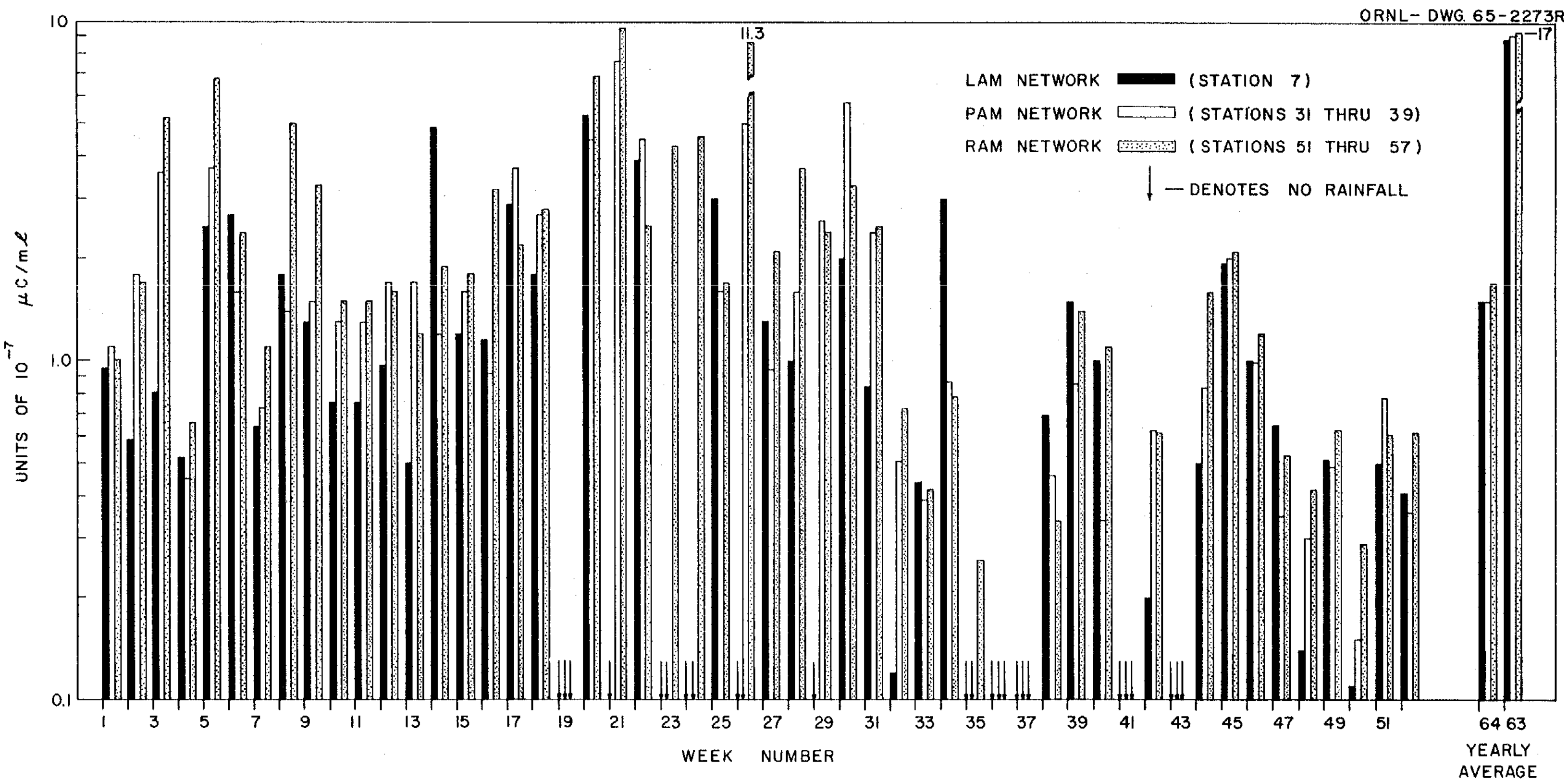

Fig. 10 Concentration of Radioactive Materials in Rainwater - 1964

Table 4 LIQUID WASTES DISCHARGED FROM WHITE OAK CREEK - 1964

	Curies		% Deviation from 1963 Weekly Average
	Total for Year	Weekly Average	
Beta Activity	234	4.5	- 50
Transuranic Alpha Emitters	0.076	0.0014	- 56

Table 5 YEARLY DISCHARGES OF RADIONUCLIDES TO CLINCH RIVER (CURIES)

Year	Gross Beta	^{137}Cs	^{106}Ru	^{90}Sr	TRE (-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co
1949	718	77	110	150	77	18	180	22	77	
1950	191	19	23	38	30		15	42	19	
1951	101	20	18	29	11		4.5	2.2	18	
1952	214	9.9	15	72	26	23	19	18	20	
1953	304	6.4	26	130	110	6.7	7.6	3.6	2.1	
1954	384	22	11	140	160	24	14	9.2	3.5	
1955	437	63	31	93	150	85	5.2	5.7	7.0	6.6
1956	582	170	29	100	140	59	12	15	3.5	46
1957	397	89	60	83	110	13	23	7.1	1.2	4.8
1958	544	55	42	150	240	30	6.0	6.0	8.2	8.7
1959	937	76	520	60	94	48	27	30	0.5	77
1960	2190	31	1900	28	48	27	38	45	5.3	72
1961	2230	15	2000	22	24	4.2	20	70	3.7	31
1962	1440	5.6	1400	9.4	11	1.2	2.2	7.7	0.36	14
1963	470	3.5	430	7.8	9.4	1.5	0.34	0.71	0.44	14
1964	234	6.0	191	6.6	13	0.3	0.16	0.07	0.29	15

Table 6 RADIOACTIVITY IN CLINCH RIVER - 1964

Location	Concentration of Radionuclides of Primary Concern in Units of 10^{-8} μc/ml						Average Concentration of Total Radioactivity	$(MPC)_w$[a]	% of $(MPC)_w$
	^{90}Sr	^{144}Ce	^{137}Cs	$^{103-106}Ru$	^{60}Co	^{95}Zr-^{95}Nb	10^{-8} μc/ml	10^{-6} μc/ml	
CRM 41.5[b]	0.12	0.08	0.04	0.31	*	0.005	0.55	1.27	0.43
CRM 20.8[c]	0.14	0.07	0.10	2.5	0.30	0.007	13	5.86	2.3
CRM 4.5[b]	0.28	0.10	0.33	4.5	0.43	0.025	5.7	4.00	1.4

[a] Weighted average $(MPC)_w$ calculated for the mixture, using $(MPC)_w$ values for specific radionuclides specified by AEC Manual, Chapter 0524, Appendix, Annex 1, Table II.

[b] Measured values.

[c] Values given for this location are calculated values based on the levels of waste released and the dilution afforded by the river; they do not include amounts of radioactive material (e.g., fall-out) that may enter the river upstream from CRM 20.8.

*None detected.

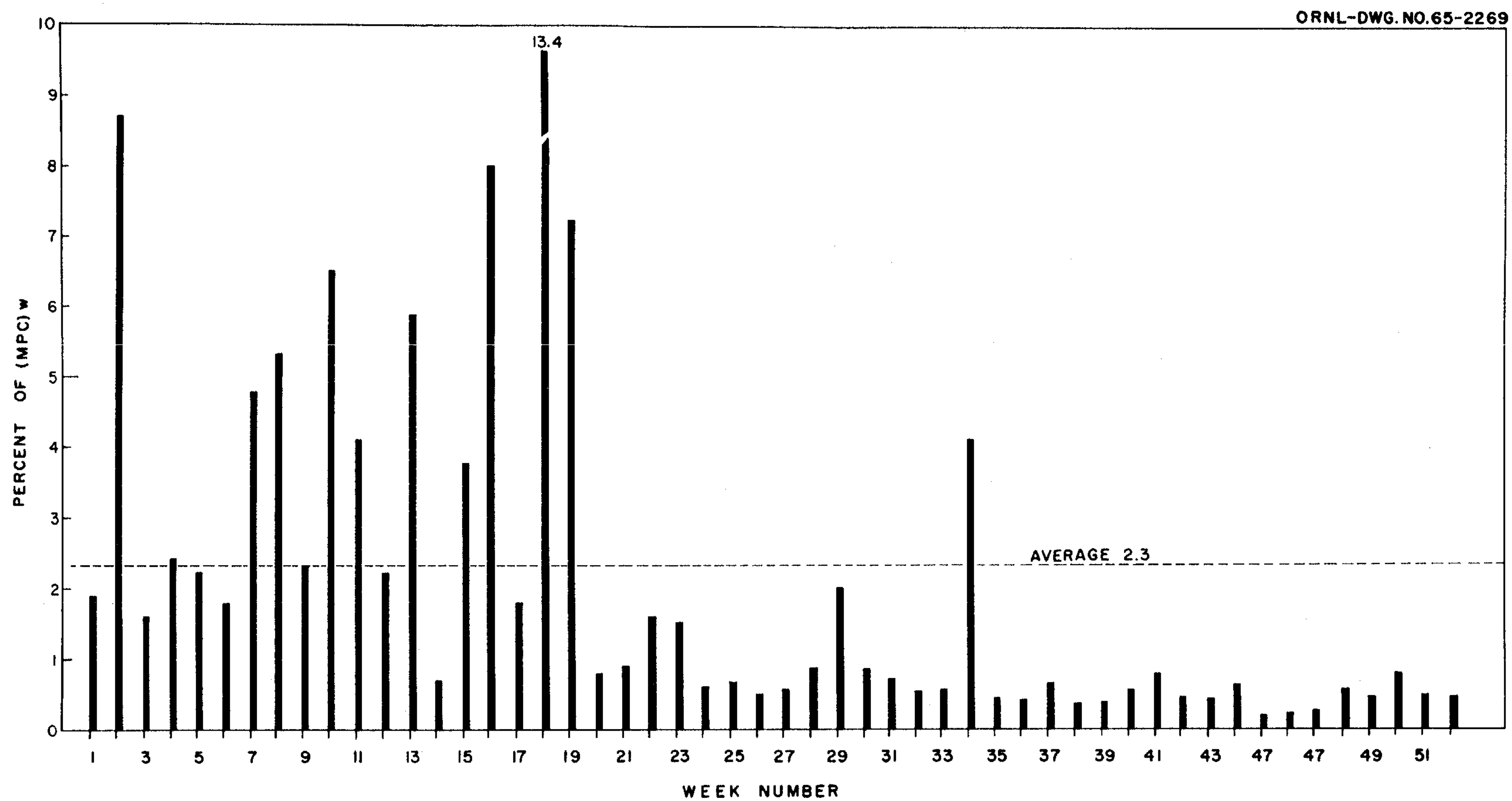

Fig. 11 Estimated Percent $(MPC)_W$ of Radioactivity in Clinch River Water Below the Mouth of White Oak Creek - 1964 (High value during week 18 was the result of low flow in the Clinch River.)

5.2.3 Potable Water - The average concentrations of ^{90}Sr in potable water at ORNL during 1964 were as follows:

Quarter Number	Concentration ^{90}Sr ($\mu c/ml$)
1	1.35×10^{-9}
2	1.44×10^{-9}
3	1.15×10^{-9}
4	1.13×10^{-9}
Average for Year	1.29×10^{-9}

The average value 1.29×10^{-9} $\mu c/ml$ represents 0.43 percent of the $(MPC)_w$ as applied to persons residing in the neighborhood of an atomic energy installation.

Based on gamma spectrometric analyses, no long-lived gamma emitting radionuclides were detected in ORNL potable water during 1964.

5.3 Milk Analyses

Two new raw milk sampling stations were added to the milk sampling networks (Hardin Valley and White Wing Road) to provide additional coverage of Laboratory releases of radioactive iodine.

The average concentration of ^{90}Sr in raw milk samples collected within a 12-mile radius of the Laboratory during 1964 was 19.5 pc/l. The average concentration of ^{90}Sr in samples collected between 12 miles and 50 miles from the Laboratory was 22.6 pc/l. These results would indicate that the ^{90}Sr content of milk in the Oak Ridge area is largely the result of fallout from previous world wide weapons tests. Figure 12 presents the weekly average concentration of ^{90}Sr in raw milk sampled from the immediate environs of Oak Ridge and shows that the peak concentration of ^{90}Sr in raw milk occurred in the spring and early summer months. (It was expected that delayed fallout from weapons testing would reach a peak in the northern hemisphere during the spring and summer months.)

The average concentration of ^{131}I in raw milk samples collected within a 12-mile radius of the Laboratory during 1964 was 11.4 pc/l. Figure 13 presents the weekly average concentrations of ^{131}I in raw milk collected at these stations compared with the weekly discharges of ^{131}I from the ORNL stacks. The correlation with the stack discharges is apparent. However, it should be noted that the yearly average concentration is near the lower limit of FRC Range II daily intake guide for ^{131}I, if one assumes an intake of 1 liter of milk per day, and that at no time during the year did the weekly average concentration exceed the upper limit of FRC Range II.

5.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each

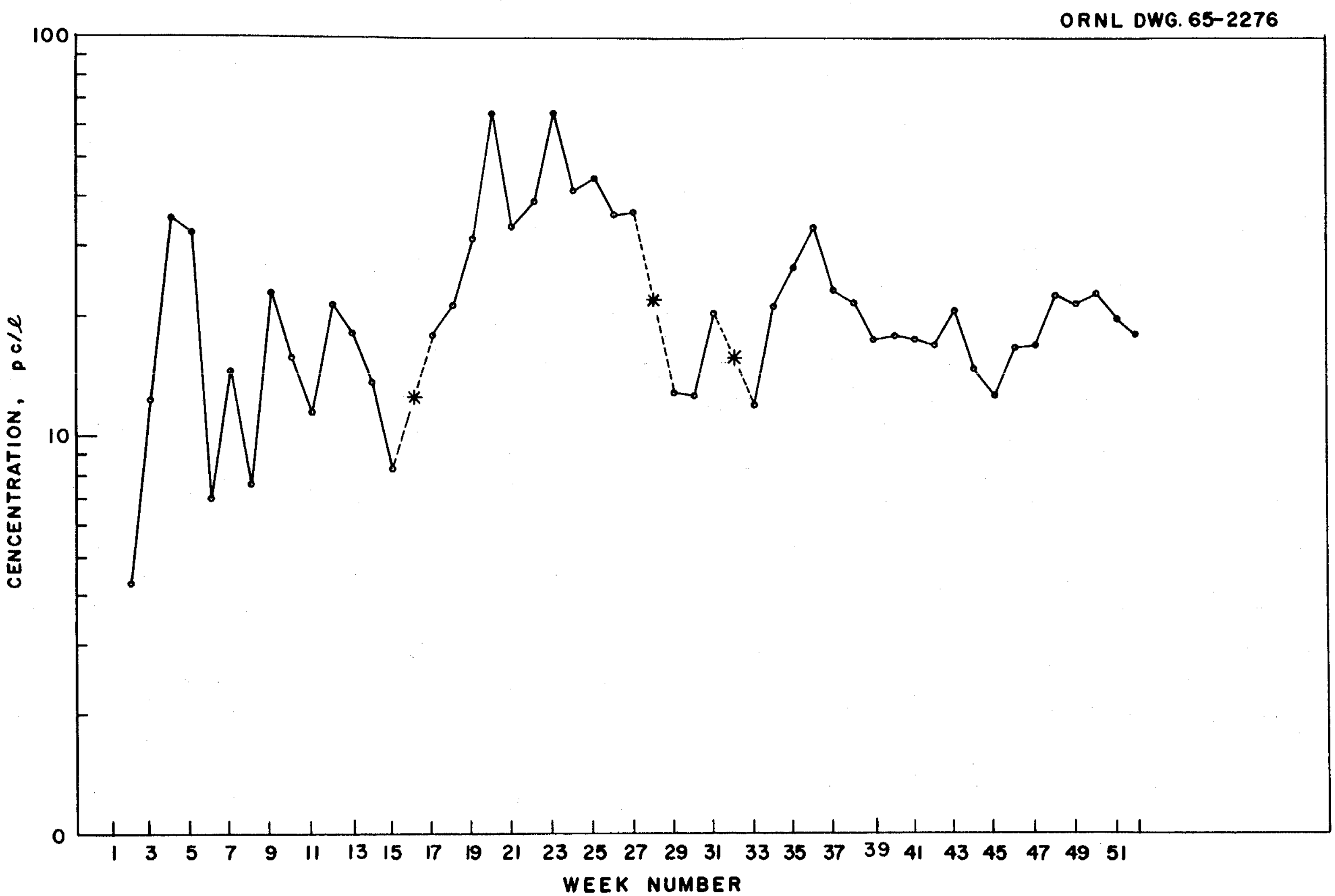

Fig. 12 Weekly Average Concentration of ^{90}Sr in Raw Milk in the Immediate Environs of Oak Ridge.

ORNL-DWG. 65-2271

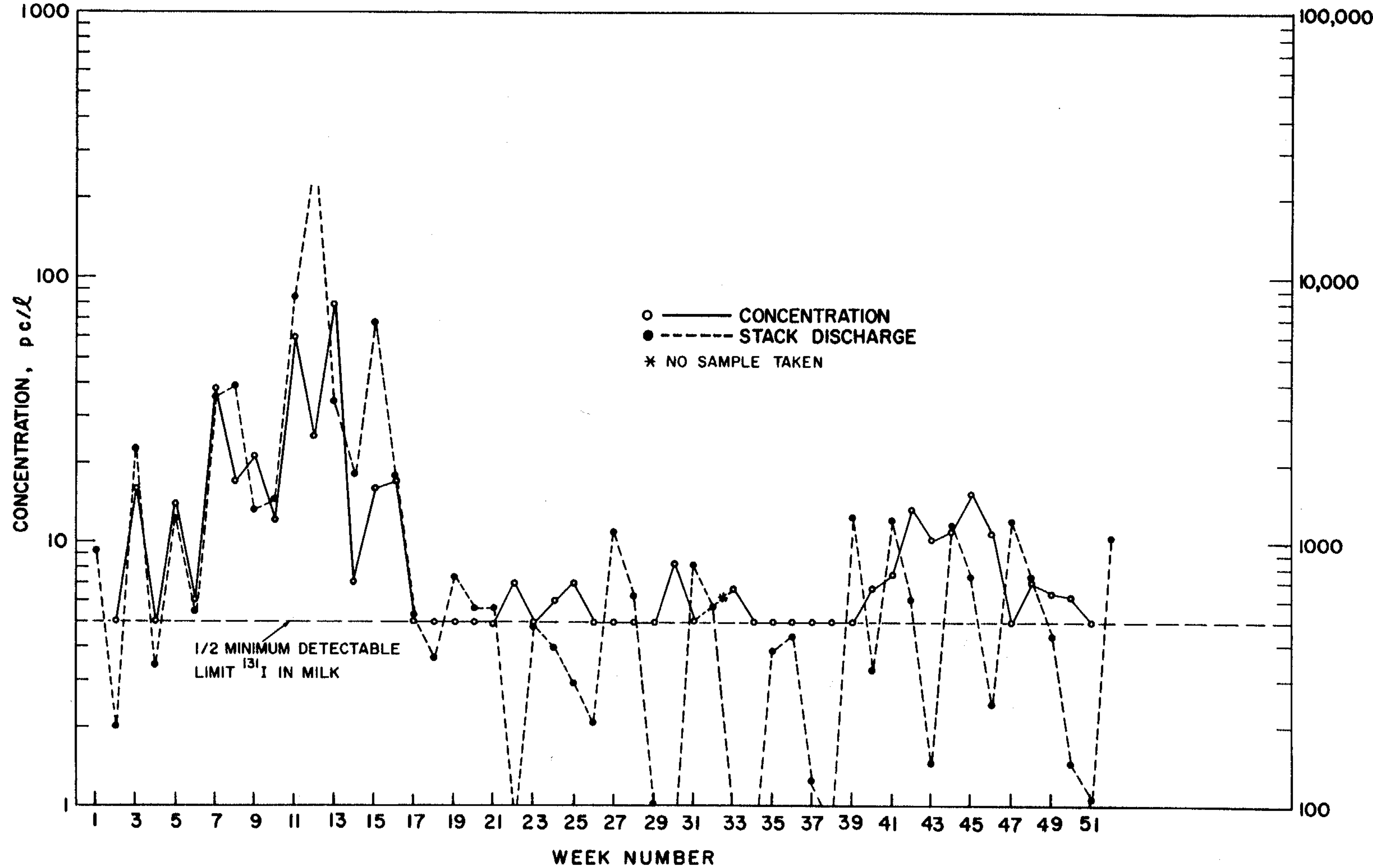

Fig. 13 Weekly Average Concentration of ^{131}I in Raw Milk Sampled from the Immediate Environs of Oak Ridge Compared with ^{131}I Discharges from ORNL Stacks - 1964.

location on a frequency of once each five weeks. The average background level during 1964 as measured at these stations was 0.014 mR/hr. Average background readings and the location of each station are presented in Figure 14.

Background measurements made monthly with a calibrated GM monitor at five selected locations adjacent to the ORNL area yielded an average background reading of 0.014 mR/hr during 1964. Corresponding measurements made at 53 locations on the ORNL site gave an average background of 0.081 mR/hr. The average background level measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite Reactor was 0.012 mR/hr. A comparison of average background values taken both on and off the X-10 site for the years 1950-64 is presented in Figure 15.

5.5 Annual Survey of the Clinch and Tennessee Rivers

The annual survey of the Clinch and Tennessee Rivers was carried out by the Applied Health Physics Section during the summer of 1964. The survey of the Clinch River extended from the mouth of the river upstream to CRM 42.8. The expanse of the Tennessee River covered by the survey extended from Fort Loudoun Reservoir at Tennessee River Mile (TRM) 604.4 through Guntersville Reservoir at TRM 354.4. The techniques and procedures used are described in ORNL 2847, "Radioactivity in Silt of the Clinch and Tennessee Rivers".

The 1964 survey showed the dispersal pattern of radioactive silt in the Clinch River to be essentially the same as in 1963 but the levels of radioactivity measured were smaller (Figure 16). The average of all gamma measurements taken in the Clinch, downstream from CRM 21.5, showed a decrease from 47 c/m observed in 1963 to 40 c/m in 1964 (Figure 17). The total number of curies discharged to the Clinch River decreased from 794 during the 12-month period just prior to the 1963 survey to 613 curies during the corresponding period prior to the 1964 survey.

The gamma count rate at the surface of the silt in Melton Hill Reservoir (upstream from the mouth of White Oak Creek) showed a decrease in 1964 when compared to 1963 (Figure 18). Many of the measurements made in the 1963 survey of Melton Hill Reservoir were made on recently inundated pastureland and was covered with decayed grass and other vegetation. The radioactivity, from fallout, contained in these organic materials resulted in higher readings in 1963 than would normally have been expected.

The average gamma count rate on bottom silt located in the Tennessee River showed a decrease in all reservoirs surveyed in 1964 (Figure 19). The average gamma count rate observed in both Clinch and Tennessee River bottom silt for the years 1951 through 1964 is given in Figure 17.

No significant changes were noted in the average concentrations of those radionuclides which comprise the major portion of the radioactivity in the Clinch and Tennessee River silt.

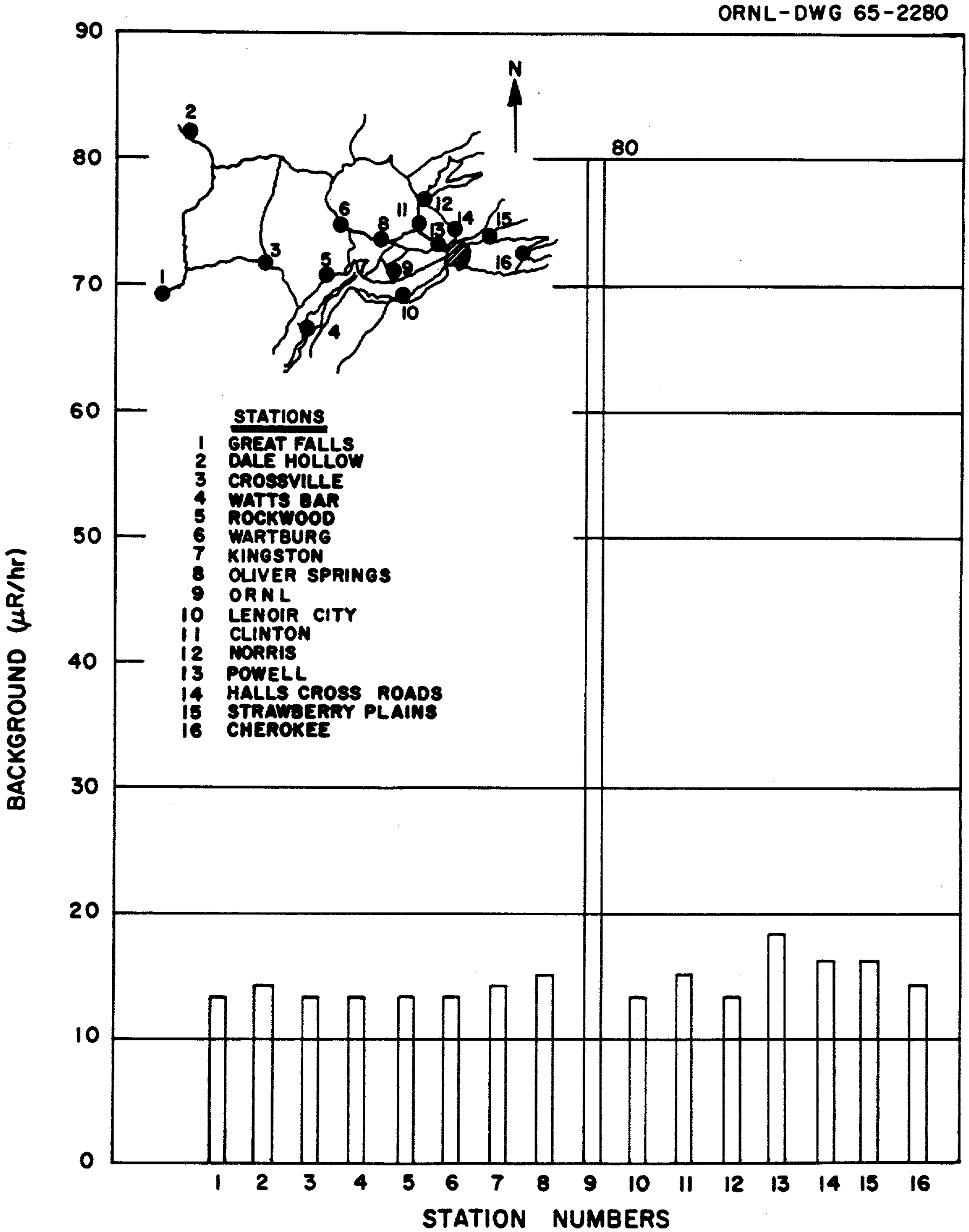

Fig. 14 Radiation Measurements Taken During 1964, 3 ft. above the Ground Surface out to Distances of 75 Miles from ORNL.

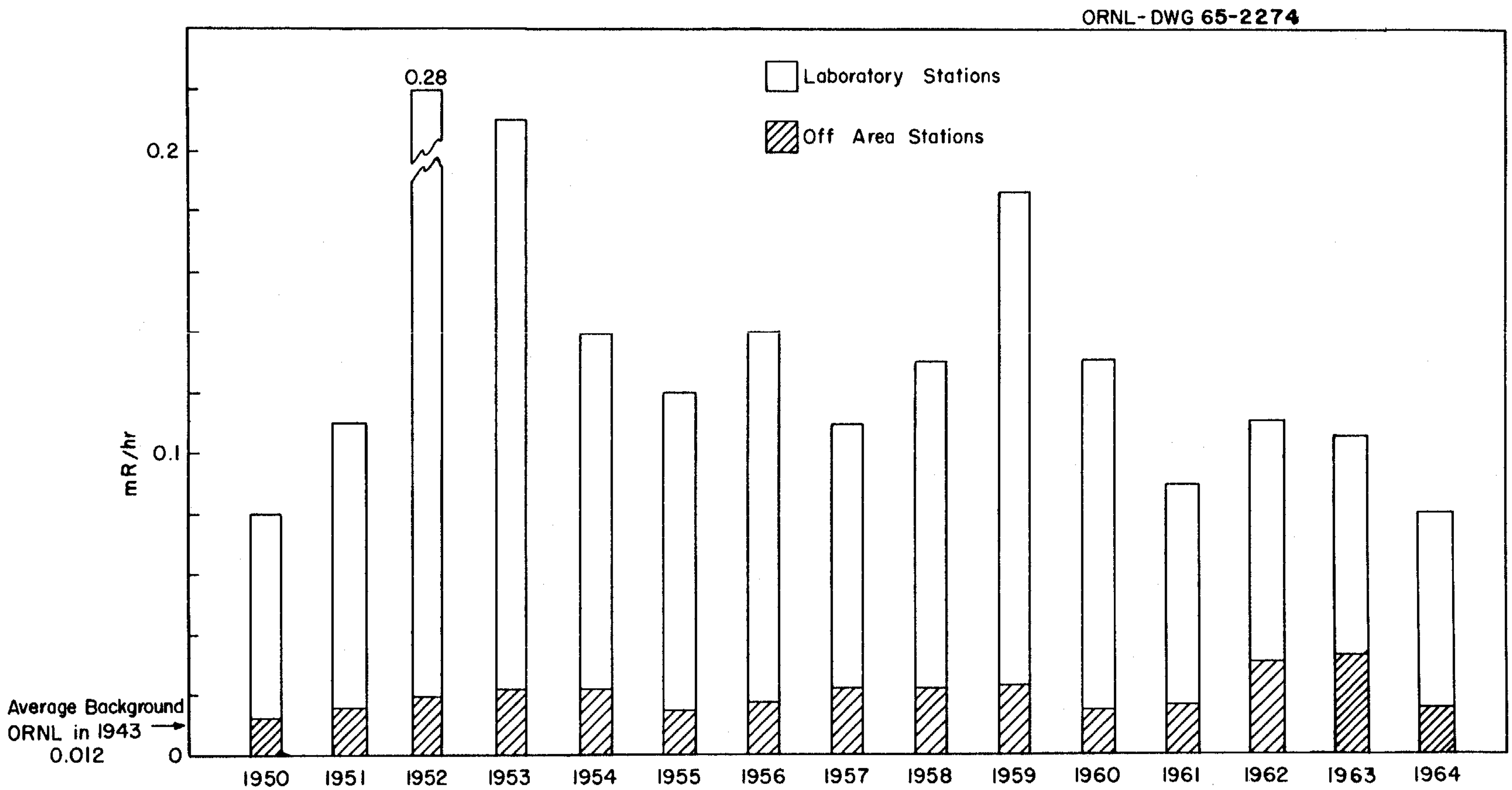

Fig. 15 Radiation Measurements Taken 3 ft above the Ground Surfaces at ORNL Compared with Like Measurements Taken Elsewhere within the AEC Controlled Area for the Years 1950-1964.

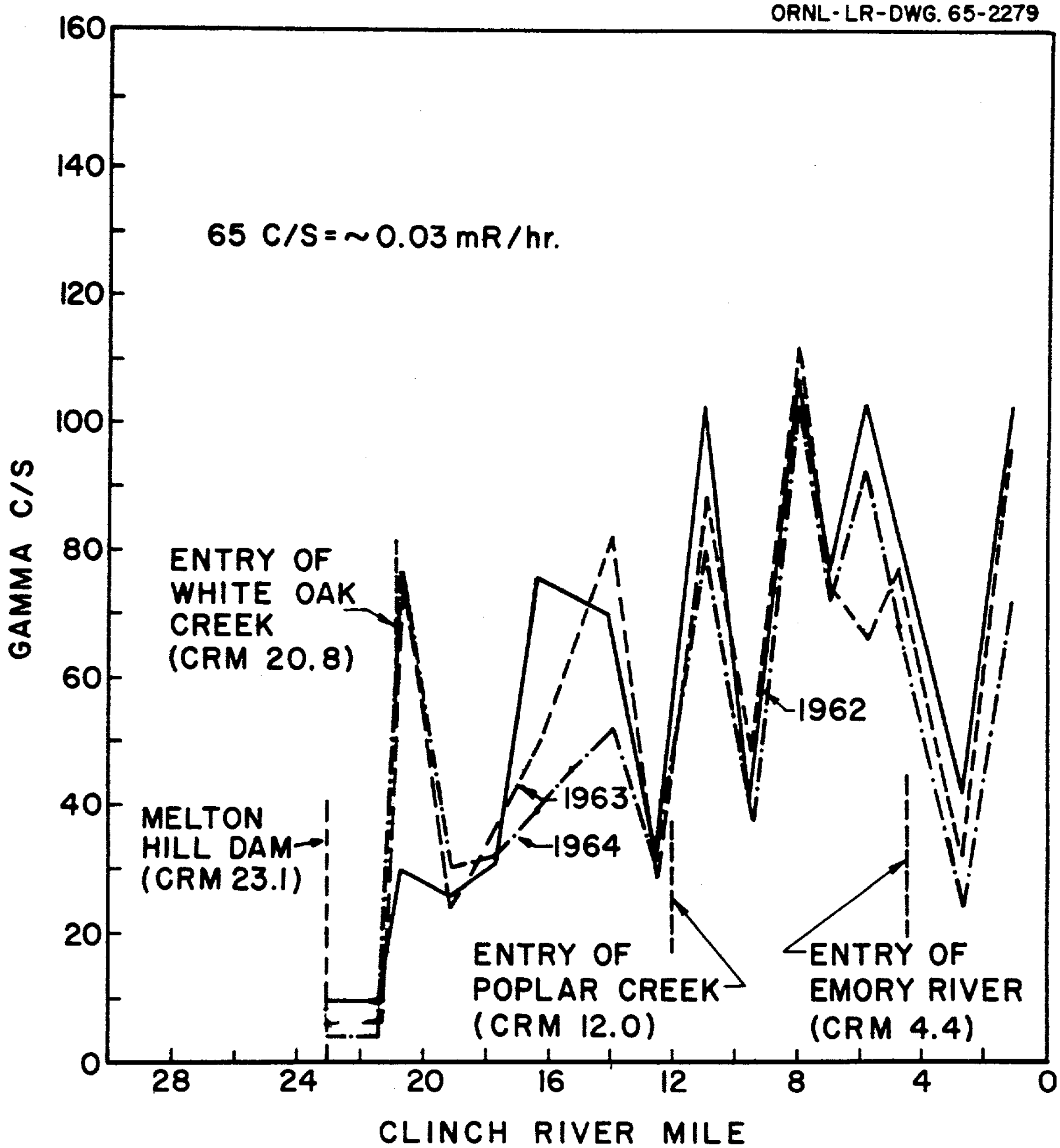

Fig. 16 Gamma Count at Surface of Clinch River Silt.

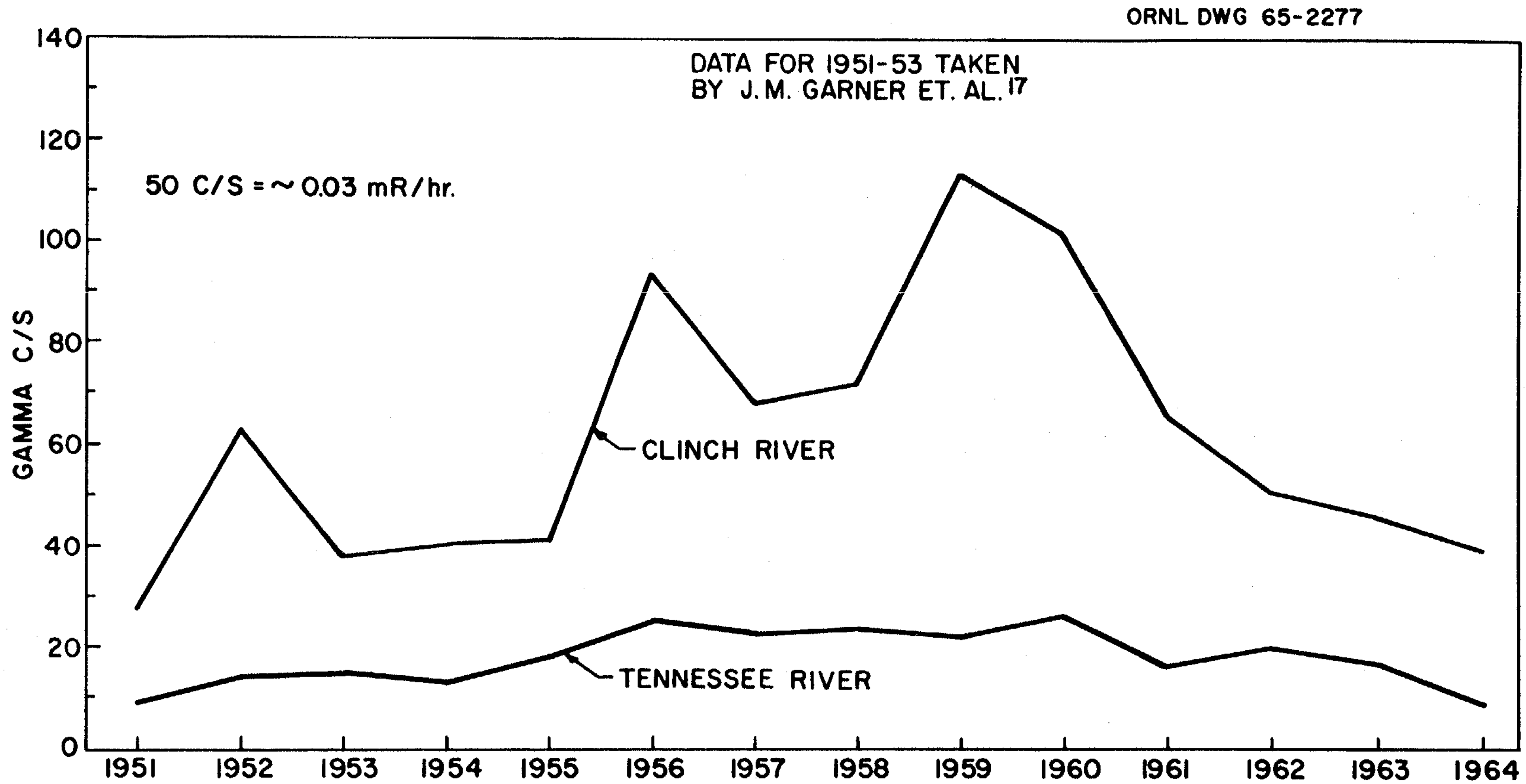

Fig. 17 Average Gamma Count at Surface of Silt, Clinch and Tennessee Rivers - 1951-1964.

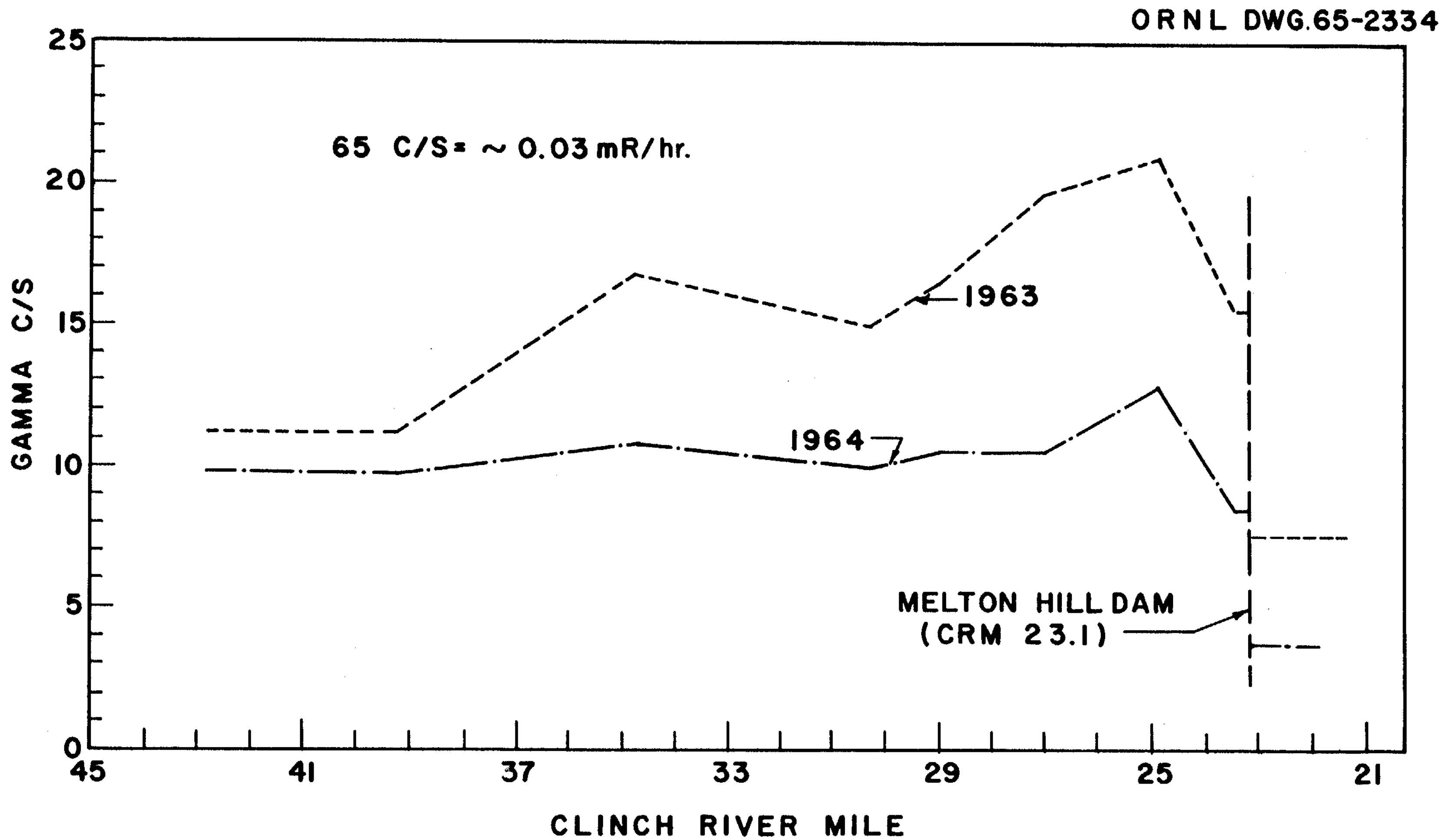

Fig. 18 Gamma Count at Surface of Clinch River Silt - Melton Hill Reservoir.

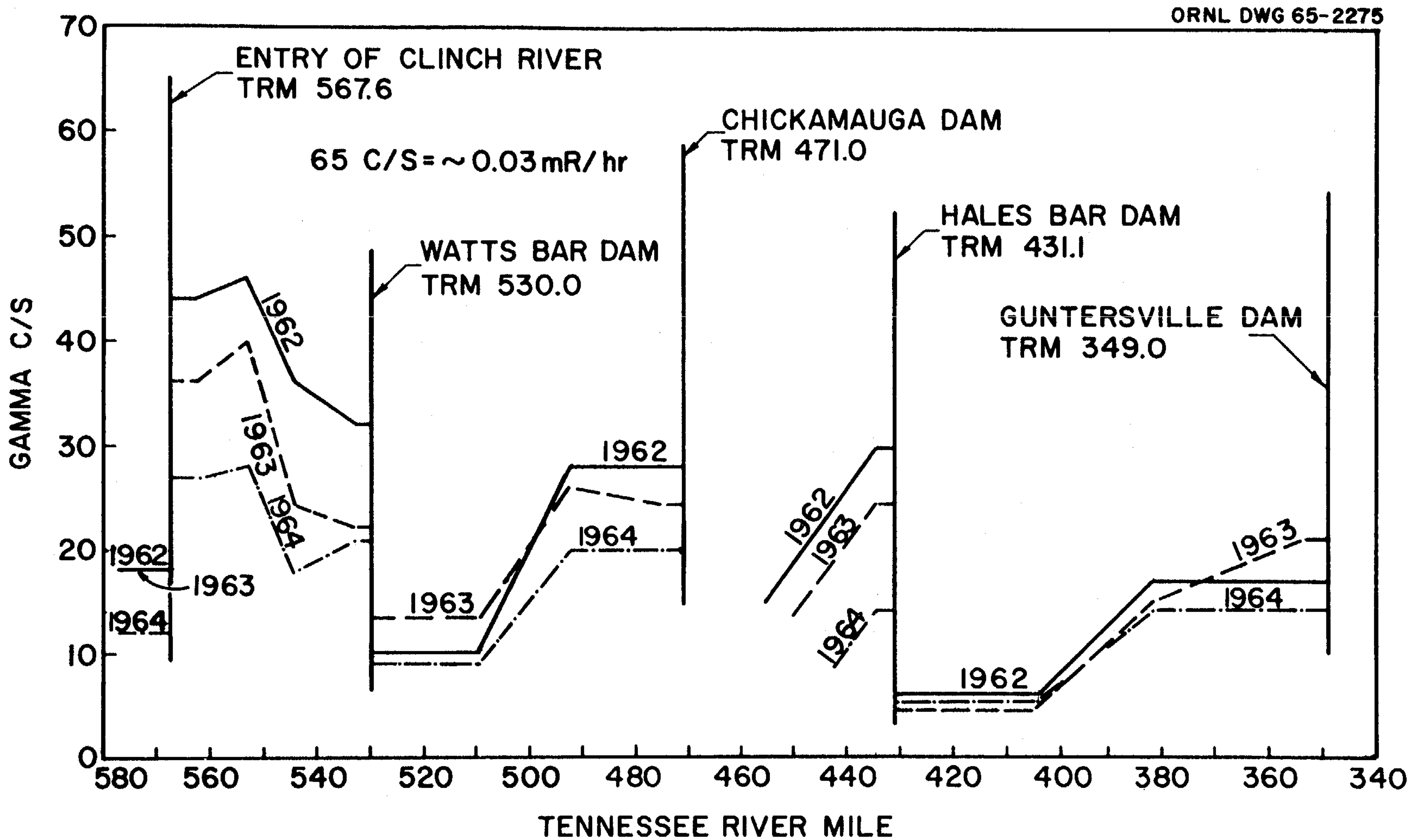

Fig. 19 Gamma Count at Surface of Tennessee River Silt.

Radiochemical analyses data obtained from the Clinch and Tennessee River silt collected in the 1963 and 1964 surveys are given in Tables 7 and 8.

Decreases in concentrations of ^{90}Sr, ^{95}Zr + ^{95}Nb and the rare earths were found in the silt of both the Clinch and Tennessee Rivers as well as in Fort Loudoun "background" silt. These decreases are apparently due to a decrease in world wide fallout from weapons tests.

Table 7 RADIOUCLIDES IN CLINCH RIVER SILT - 1963-1964
(Units of 10^{-6} μc/g of Dried Silt)

Location	^{137}Cs		^{144}Ce		^{90}Sr		^{60}Co		$^{103-106}Ru$		$^{95}Zr + ^{95}Nb$		TRE[a] + ^{90}Y (as ^{90}Y)	
	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964
CRM 42.8	2.6	8.7	17	21	0.29	0.45	*	*	14	16	14	*	15	9.0
39.1	1.3	4.3	7.7	10	0.49	0.45	*	*	6.4	7.7	6.3	*	6.3	3.2
34.7	2.6	7.5	19	19	0.34	0.72	*	*	14	16	16	*	16	7.2
31.1	2.7	6.7	16	18	0.36	0.50	*	*	13	13	12	*	9	5.0
29.0	2.7	4.0	16	9.3	0.52	0.61	*	*	12	7.5	13	*	9	5.4
27.0	3.0	6.2	22	16	0.50	0.50	*	*	16	11	19	*	15	5.4
24.9	3.0	5.1	20	14	0.52	0.45	*	*	15	9.9	17	*	14	4.5
23.4	0.90	1.5	3.8	3.5	0.43	0.50	*	*	2.8	2.1	3.4	*	3.3	1.5
Average	2.4	5.5	14	14	0.43	0.52			12	10	13		11	5.1
CRM 21.5	2.7	0.68	0.43	1.4	0.63	0.29	*	0.36	0.45	0.99	0.14	*	0.89	*
19.1	2.9	69	0.90	6.6	0.74	0.72	1.9	5.1	4.4	15	0.90	0.68	5.2	3.2
16.3	218	172	4.2	5.6	3.1	1.4	16	11	17	12	2.2	*	34	6.0
15.2	16	71	3.5	5.3	0.81	0.72	2.8	5.4	13	11	3.8	*	5.6	7.4
14.0	150	62	5.0	6.8	1.7	0.50	12	6.2	29	20	3.4	*	23	3.2
11.0	75	77	8.8	13	1.2	0.77	8.0	8.9	35	34	7.4	*	22	4.3
8.0	62	105	8.8	14	0.90	0.88	8.9	12	48	43	6.9	*	16	7.9
5.8	67	89	12	14	1.3	0.83	9.5	11	49	40	10	*	19	6.4
4.7	53	100	11	15	1.4	0.86	8.5	11	44	41	9.7	*	17	8.2
2.6	63	48	6.7	11	0.81	0.54	7.7	5.6	26	21	5.4	*	17	2.8
1.1	68	67	17	21	1.4	0.59	8.5	7.8	50	37	17	*	23	6.8
Average	71	78	7.1	10	1.3	0.74	8.4	7.7	29	25	6.1		17	5.1

[a]Total Rare Earths minus cerium
*None detected

Table 8 RADIONUCLIDES IN TENNESSEE RIVER SILT - 1963-1964
(Units of 10^{-6} μc/g of Dried Silt)

Location	^{137}Cs		^{144}Ce		^{90}Sr		^{60}Co		$^{103-106}Ru$		^{95}Zr + ^{95}Nb		TRE[a] + ^{90}Y (as ^{90}Y)	
	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964	1963	1964
TRM 570.8	1.8	2.8	7.2	9.9	0.61	0.23	0.45	*	7.5	6.3	6.3	0.18	4.1	2.1
562.7	18	19	12	15	0.77	0.32	2.7	2.1	21	15	9.8	0.36	9.2	2.2
552.7	26	26	14	9.8	0.65	0.88	3.4	3.7	26	19	12	*	13	6.6
543.8	14	20	11	9.6	0.52	0.54	2.3	3.2	14	17	11	*	11	6.6
532.0	13	16	10	7.7	0.56	0.74	2.3	2.9	13	15	9.5	*	8.8	4.8
491.9	10	13	10	9.2	0.52	0.61	1.8	2.3	10	16	8.1	*	9.8	5.2
475.1	9.9	12	11	15	0.74	0.16	1.8	1.5	9.9	13	9.0	*	8.9	*
434.1	8.1	8.2	20	21	0.23	0.34	0.90	0.45	8.1	15	22	0.36	19	*
381.2	5.9	7.3	9.1	15	0.61	0.29	0.90	0.59	5.9	11	9.0	*	8.4	2.4
354.4	5.0	6.6	13	13	1.1	0.43	0.90	0.32	5.0	11	11	0.32	12	*
Average	10	13	12	13	0.63	0.45	1.7	1.7	12	14	11	0.12	10	3.0
Fort Loudoun Background Data														
TRM 604.4	2.2	3.3	10	8.1	0.61	0.47	*	*	9.4	5.8	6.8	0.59	5.9	2.4
615.8	2.1	--	9.5	--	0.54	--	*	--	8.5	--	5.5	--	9.5	--
Average	2.2	3.3	9.8	8.1	0.58	0.47			9.0	5.8	6.2	0.59	7.2	2.4

[a]TRE - total rare earths minus cerium

*None detected

--No samples taken in 1964

6.0 PERSONNEL MONITORING

It is the policy of Oak Ridge National Laboratory to monitor the radiation exposure of each employee and all other individuals who enter Laboratory areas where there is a potential for radiation exposure. Dose analysis is accomplished mainly through the use of personnel meters, bio-assays, and in vivo counting (whole body counter) techniques.

6.1 Dose Analysis Summary, 1964

6.1.1 External Exposures - No employee received an external radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest whole body dose received by an employee was about 4.2 rem or 35 percent of the maximum permissible annual dose. The range of doses for persons using ORNL badge-meters is shown in Table 9.

As of December 31, 1964 only one employee had a cumulative whole body dose which exceeded the recommended maximum permissible dose as based on the age proration formula 5(N - 18) (Table 10). Nearly all of the dose recorded for this individual (67.6 rem) resulted from an accidental exposure which occurred during 1957, and at the end of 1964 represented about 104 percent of the dose permitted by the age proration formula; his average dose per year of employment at ORNL was 8.3 rem. Only one other employee had an average annual exposure rate that exceeded 5 rem per year of employment (Table 11) and the average annual dose for this employee was 5.1 rem being accured over a period of about 13 years.

The highest cumulative dose to the skin of the whole body received by an employee during 1964 was about 20.7 rem or 69 percent of the maximum permissible annual skin dose of 30 rem. Only one other employee received a skin dose that exceeded 30 percent of the maximum permissible; this cumulative exposure dose was about 10.2 rem or 34 percent of the maximum permissible annual dose.

As of December 31, 1964 the highest cumulative dose of whole body radiation received by an employee was approximately 85 rem. This dose was accured over an employment period of about 20 years and represented an annual exposure of about 4.1 rem.

The highest cumulative hand exposure recorded during 1964 was about 14.4 rem or 19 percent of the recommended maximum permissible annual dose to the extremities.

During 1964, except with the foreknowledge of Health Physics, no visitor-type meter was found to have sustained a detectable radiation dose.

Table 9 Dose Data Summary for Laboratory Population Involving Exposure to Whole Body Radiation—1964

Group	Number of Rem Doses in Each Range							
	0-1	1-2	2-3	3-4	4-5	5-6	6 up	Total
ORNL Employees	5150	148	35	9	1	0	0	5343
ORNL-Badged Non-Employees	928	0	0	0	0	0	0	928
TOTAL	6078	148	35	9	1	0	0	6271

Table 10 Average Rem per Year Since Age 18

	Number of Doses in Each Range				
	0-2.5	2.5-5.0	5.0-7.5	7.5 up	Total
ORNL Employees	5335	7	1	0	5343

Table 11 Average Rem per Year of Employment at ORNL

	Number of Doses in Each Range					
	0-2.5	2.5-5.0	5.0-7.5	7.5-10.0	10.0 up	Total
ORNL Employees	5317	24	1	1	0	5343

6.1.1.1 External Dose - The average of the ten highest whole body doses of ORNL employees for each of the years 1958 through 1964 are shown in Figure 20. A downward trend in this value is apparent. The highest dose for each of those years is shown also.

The dose ranges versus the number of employees for each range for the years 1958 through 1964 are shown in Figure 21. Although the total number of employees increased slightly during the six-year period, the number of persons in the higher dose ranges decreased progressively.

The average annual dose to ORNL employees for the years 1958 through 1964 is the subject of Figure 22. This rather arbitrary quantity is obtained by dividing the sum of all doses for the year by the number of employees involved.

6.1.2 Internal Exposures - During 1964 two employees received an exposure, by inhalation, to ^{90}Sr which resulted in an estimated deposition of between 30 and 50 percent of the maximum permissible lung burden.

An employee who was exposed to ^{3}H in 1963 (see ORNL-3665, page 9) continued a work restriction with non-radioactive materials until the body burden, as indicated by excretion measurements, was reduced to the lower limit of detectability.

Three employees continued to have estimated body burdens of transuranic alpha emitters (mainly ^{239}Pu) of 35 to 40 percent of the recommended maximum permissible value.[5] Health Physics procedures require that individuals who exceed 30 percent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

During 1964 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than one-half a maximum permissible body burden.[6]

6.2 External Dose Techniques

6.2.1 Film Meters - Film meters are issued to all persons who have access to ORNL facilities in which there is a potential for radiation exposures in excess of RPG levels. Either an ORNL badge-meter (Figure 23) or a temporary pass-meter (Figure 24) may be used. Badge-meters are assigned to all ORNL employees, and to certain other persons who are authorized to enter ORNL facilities frequently. Temporary pass-meters are issued in lieu of badge-meters for short-term use.

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[5] AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 percent of a maximum permissible limit.

[6] Handbook 69 values are the basis for these determinations.

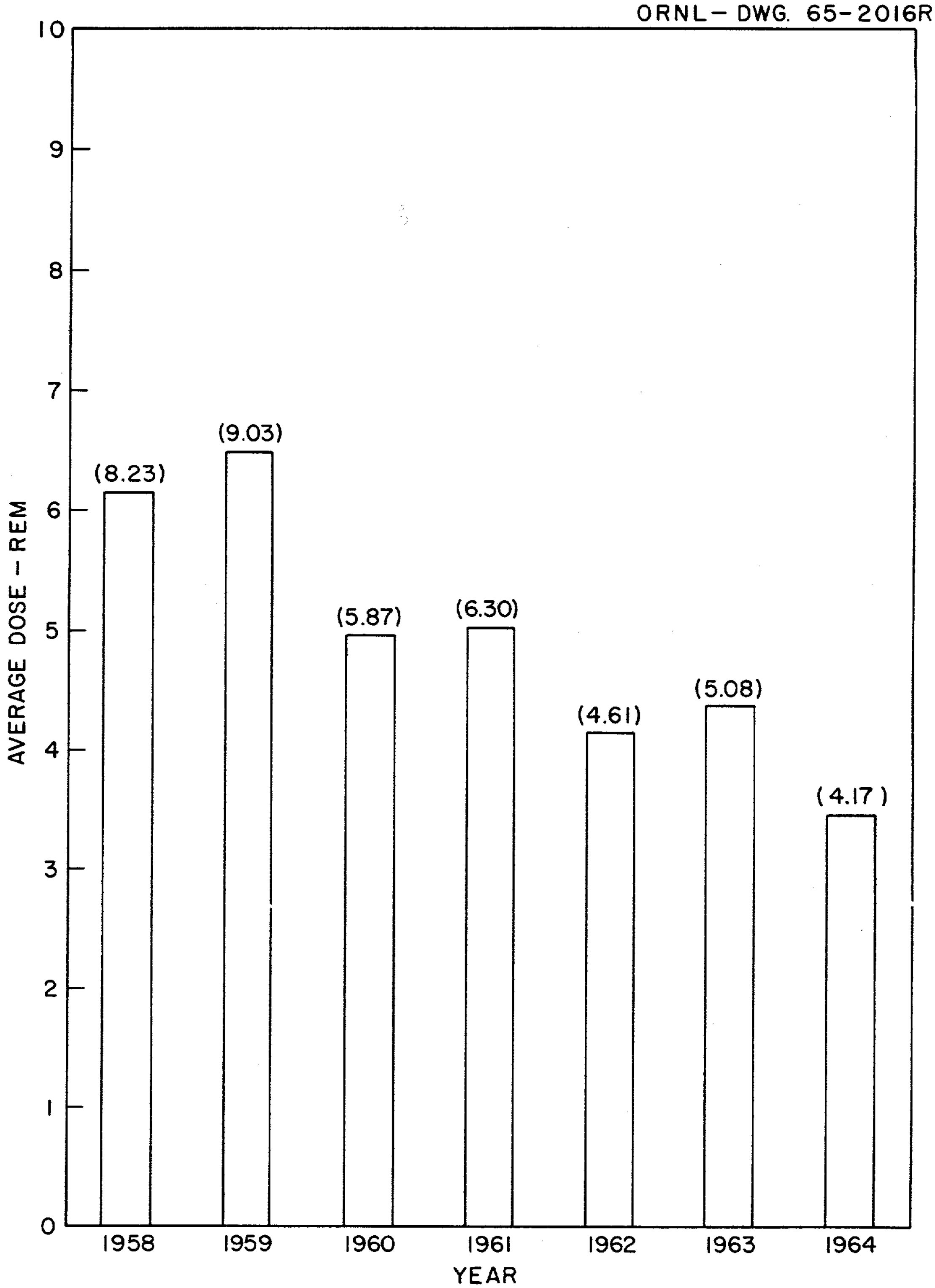

Fig. 20 Average of the Ten Highest Annual Whole Body Doses by Year (The Highest Individual Dose Shown in Parentheses)

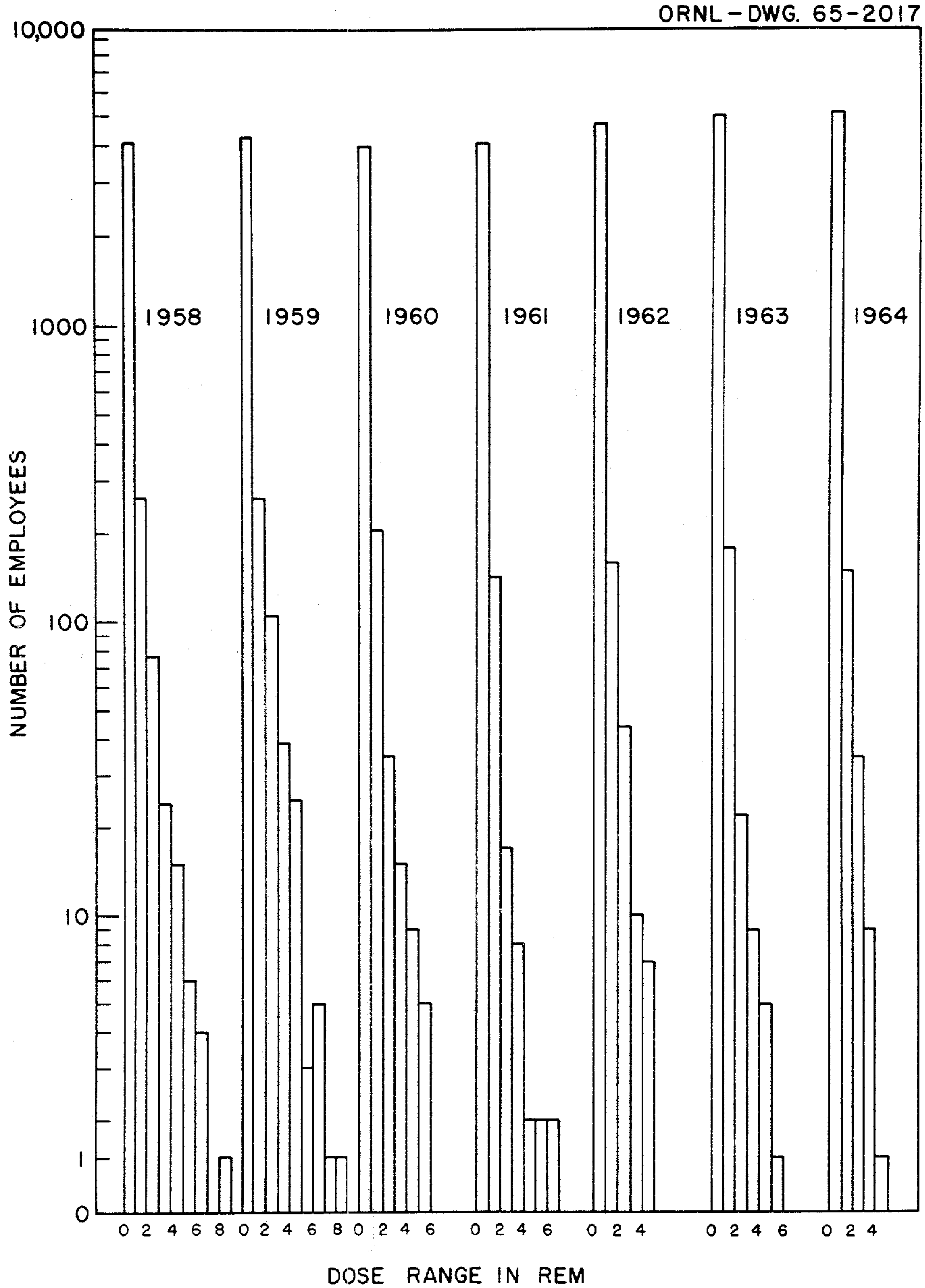

Fig. 21 Whole Body Radiation Dose Range for Employees - 1958-1964.

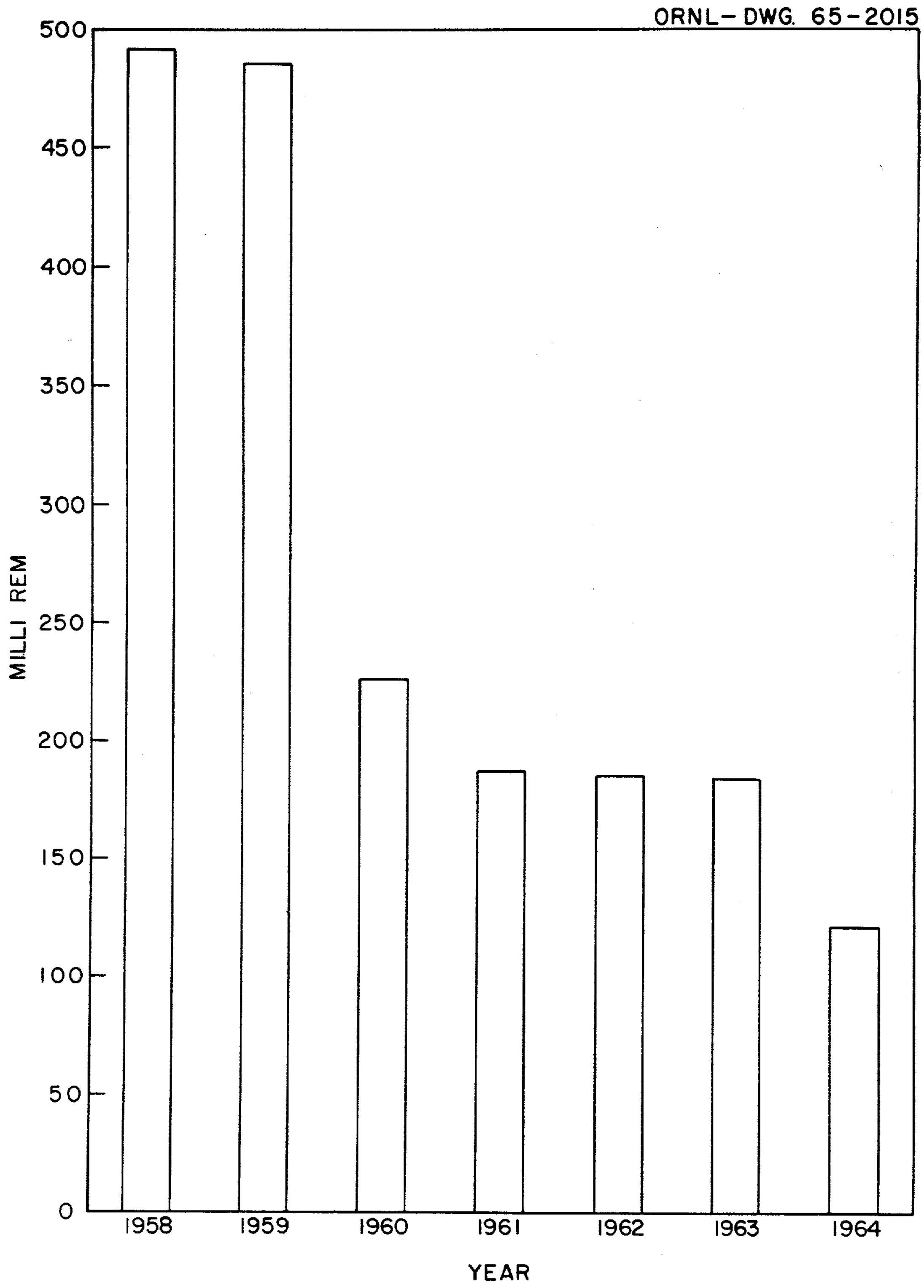

Fig. 22 Average Annual Whole Body Dose to the Average ORNL Employee.

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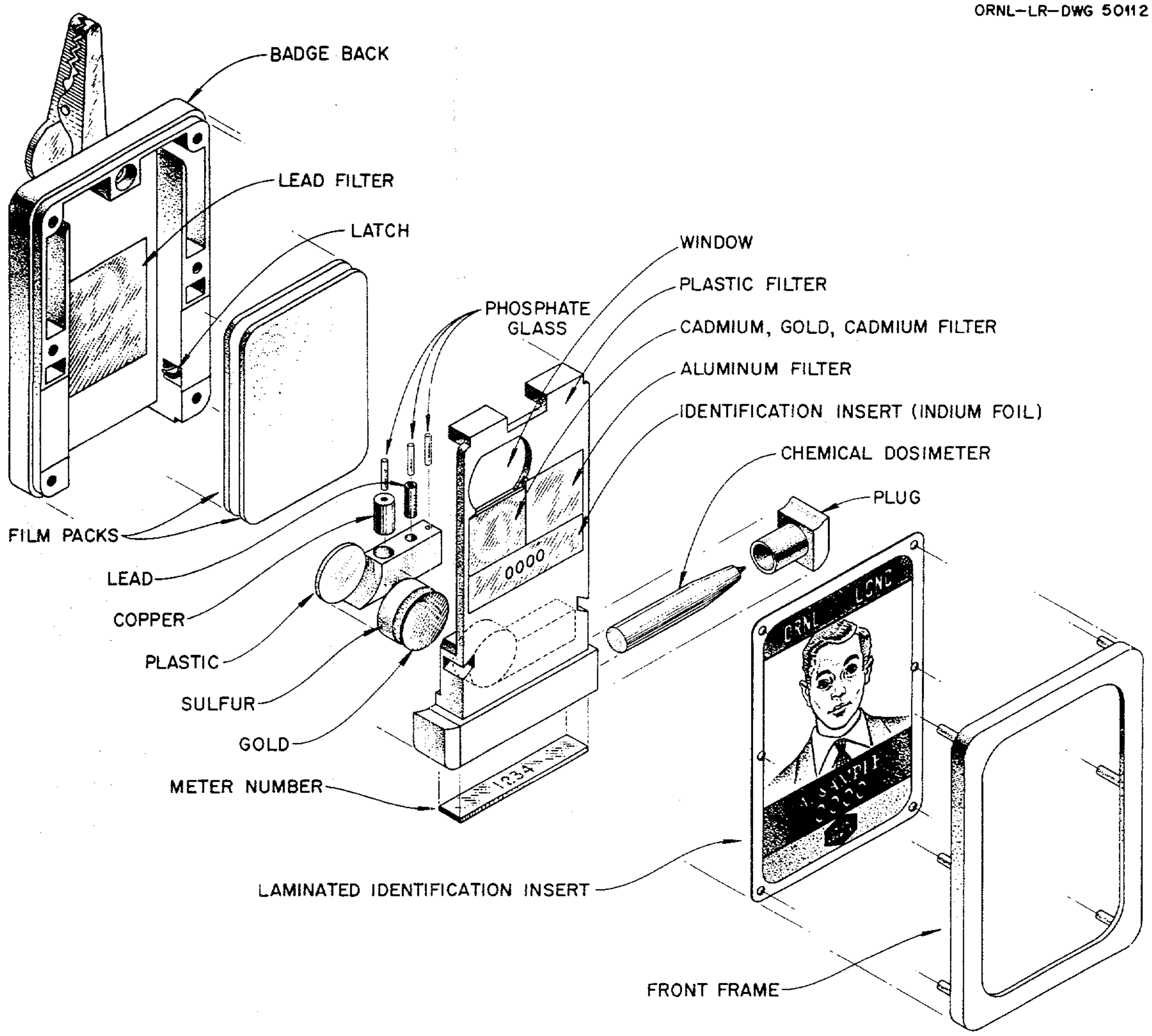

Fig. 23 ORNL Badge-Meter, Model II.

Fig. 24 Typical Temporary Security Passes Equipped with Monitoring Film.

NTA (nuclear track) film packets are included in all badge-meters issued to ORNL employees and assignees. The NTA films are processed routinely if the badge-meter is assigned to an individual who normally works where there may be exposure to neutrons. Spot checks are made on ten percent of the NTA films used by other individuals.

Beta-gamma sensitive films from badge-meters issued to all employees and assignees are processed routinely each calendar quarter (or more frequently if necessary). Films used in other meters are processed as conditions of use may require. High-level radiation dosimetry components of the badge-meters (sulfur, gold, indium, and metaphosphate glass) are for use in the event that doses exceed the capability of the monitoring films.

For each ORNL division which had one or more employees who sustained a dose greater than one rem for the year, the number of employees so exposed are displayed in Figure 25. The divisions are identified by division names. It may be noted that only 12 divisions had employees with doses greater than 1 rem, five had employees with doses greater than 2 rem, and only one had employees with doses greater than 3 rem.

6.2.2 Pocket Meters - Pocket meters (indirect reading, ionization chambers), Figure 26, are made available at all principal points of entry to ORNL premises. A pair of pocket meters is carried for the duration of a work shift by persons who work in an area where the potential for an exposure of 20 mR or more exists during the work shift. Pocket meter pairs are processed each day by health physics technicians and readings of 20 mR or more are reported daily to supervision.

Pocket meters provide for a day-to-day record of integrated exposures and warn if excessive exposures occur.

Figure 27 is a display of the comparison between whole body doses as determined from film meters and the total recorded pocket meter readings for the ten highest whole body dose cases for the year 1964.

6.2.3 Hand Exposure Meters - Hand exposure meters (Figure 28) are film-loaded finger rings used to measure hand exposure. Hand exposure meters are issued on a weekly basis to persons for use during operations where it is likely that the hand dose is such as to exceed 1 rem during the week. Hand exposure meters are issued and collected by Radiation Survey Unit personnel who determine the need for this type of monitoring and arrange for a processing schedule.

6.2.4 Metering Resume - Shown in Table 12 are the quantities of personnel metering devices used and processed during 1964. The number of films processed was less than in 1963, and should be even less in 1965, because of the reduction in monitoring requirements for visitors to the Laboratory.

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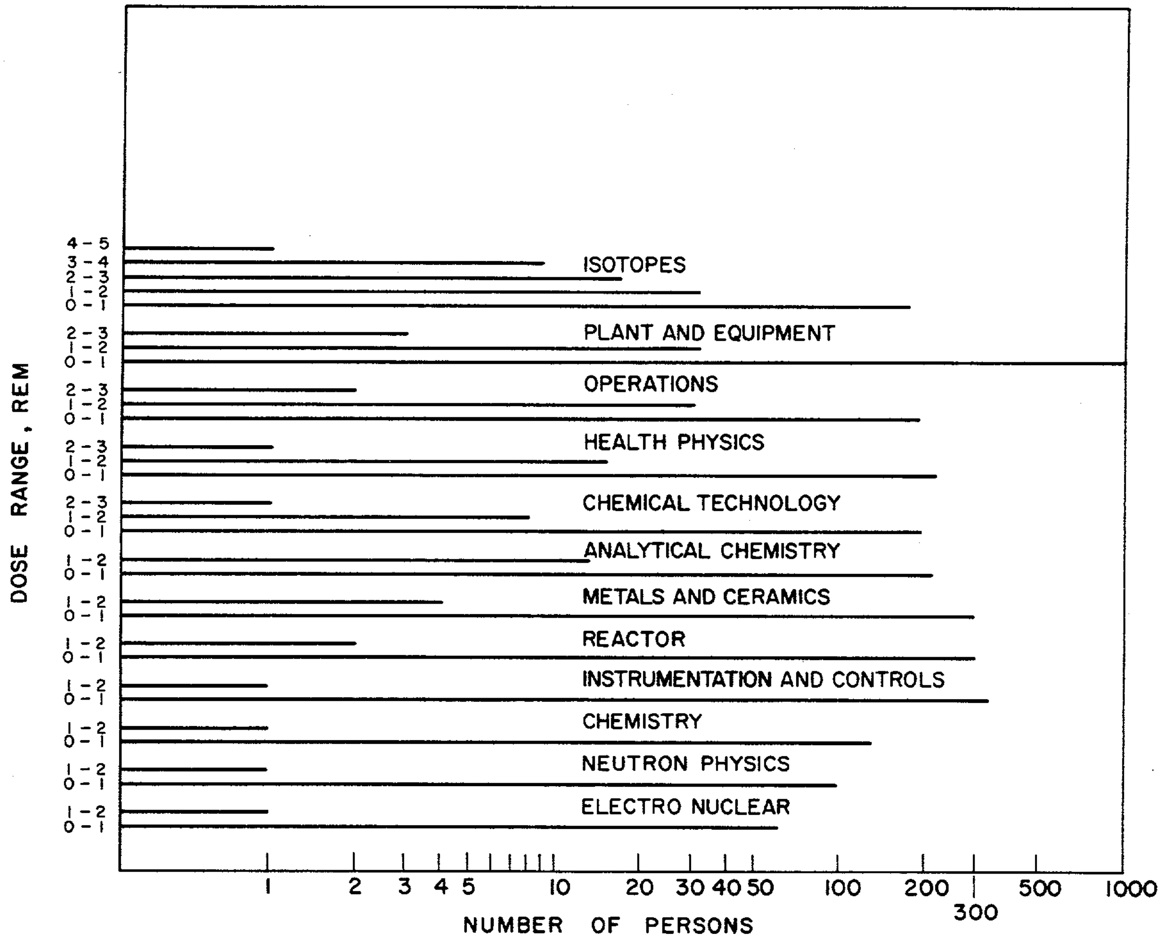

Fig. 25 Personnel Dose (Whole Body) by ORNL Division Having One or More Doses, One Rem or Greater, in 1964.

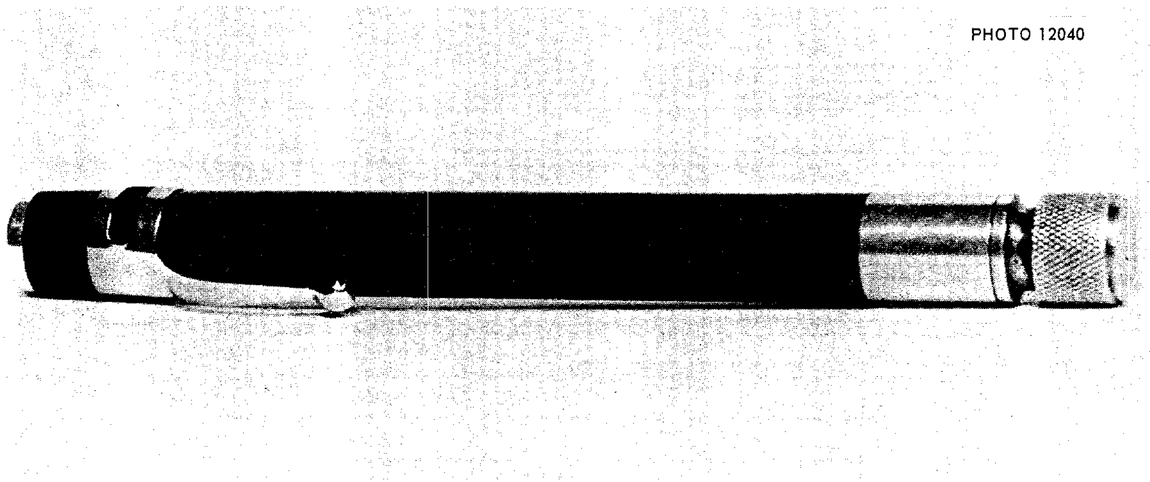

Fig. 26 Pocket Ionization Chamber.

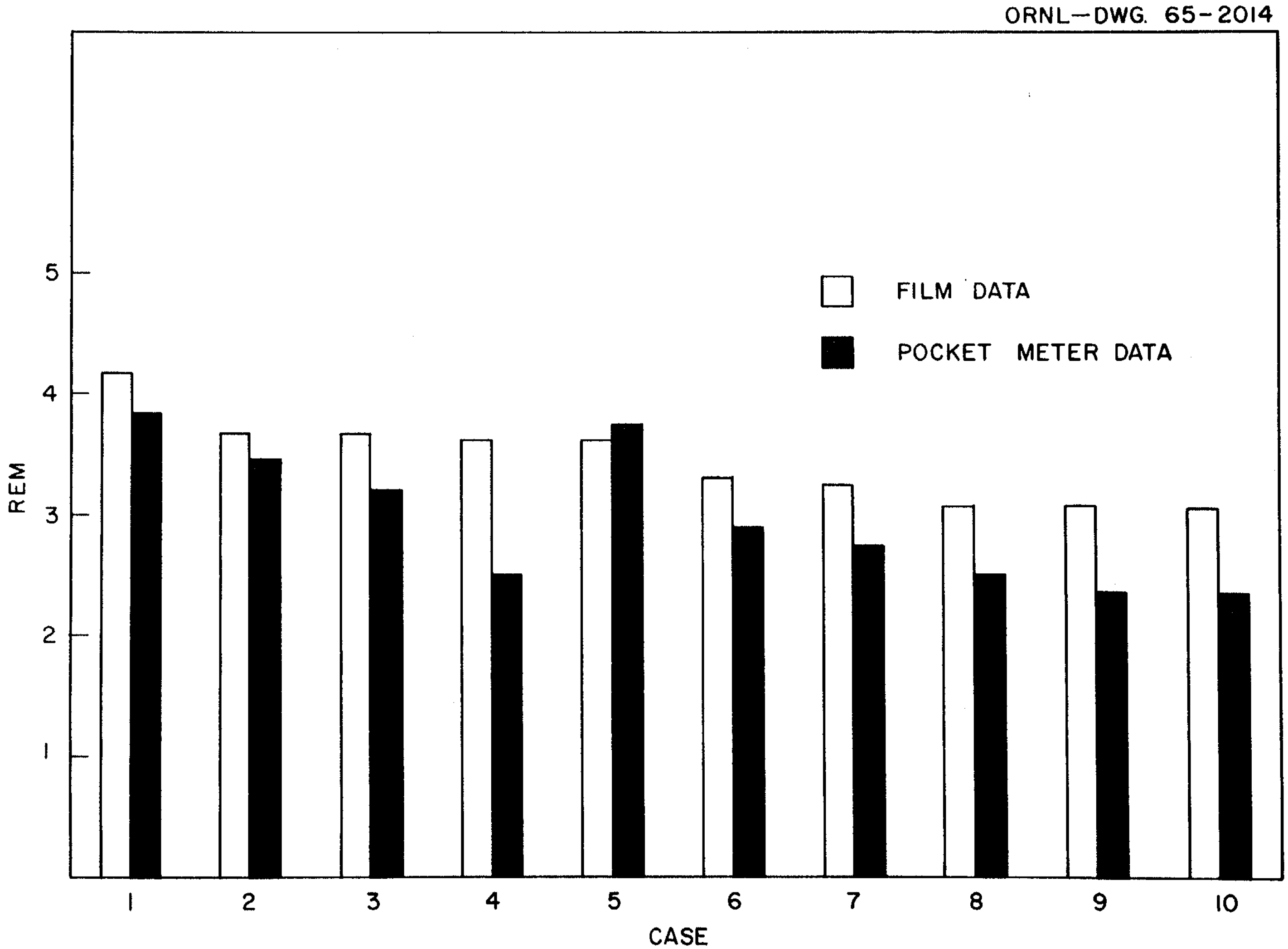

Fig. 27 The Ten Highest Whole Body Radiation Dose Cases Compared with Concurrent Pocket Meter Totals for 1964.

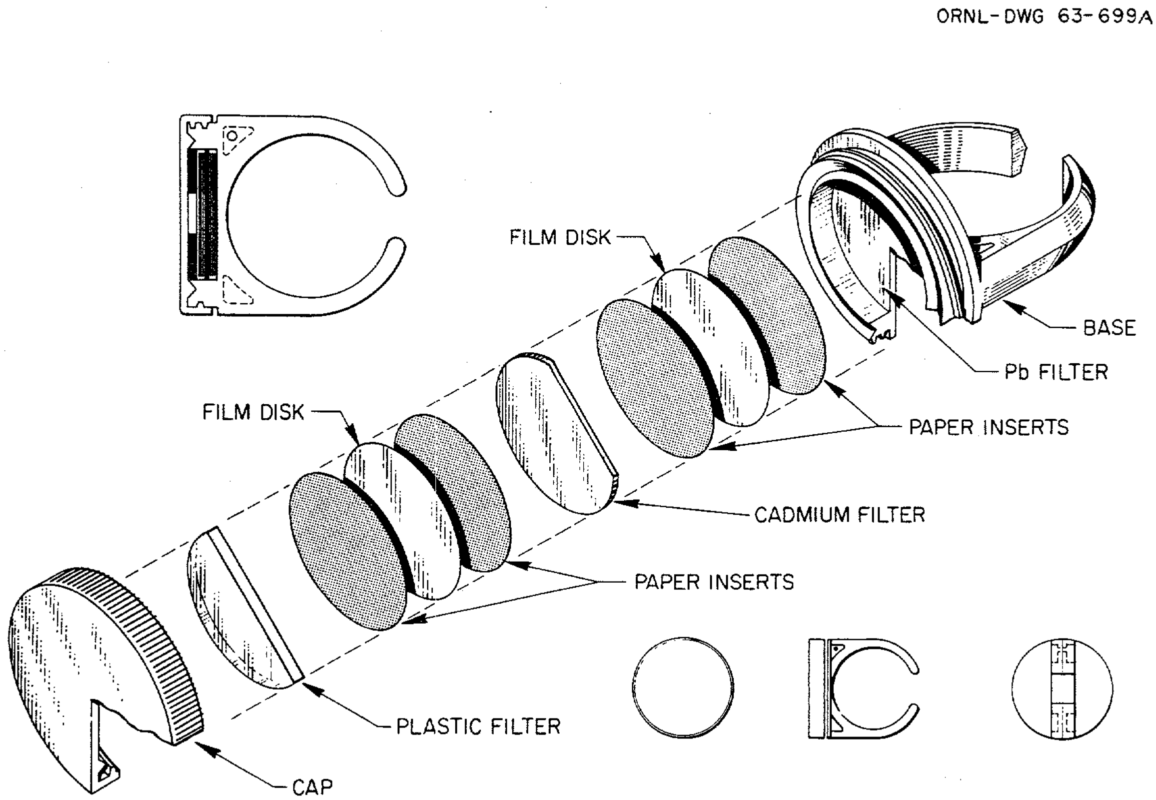

Fig. 28 Details of the ORNL Hand Exposure Meter.

Table 12 PERSONNEL METER SERVICES

			TOTAL
A. Pocket Meter Usage			
1. Number of Pairs Used	ORNL	144,026	
	H. K. Ferguson	36,928	
			180,954
2. Average Number of Users per Quarter			
	ORNL	1,305	
	H. K. Ferguson	340	
			1,645
B. Badge-Meter Usage			
1. Badge-Meter Film Changes			21,634
2. Temporary Meter Film Changes			31,877
Total Number of Meters Issued			53,511
C. Films Processed for Monitoring Data			
1. Beta-Gamma			25,244
2. NTA			2,042
3. Hand Meter			1,741
TOTAL			29,027

6.3 Internal Dose Techniques

6.3.1 Bio-Assays - Urine and fecal samples are analyzed for the purpose of making internal dose determinations. The frequency of sampling and the type of radiochemical analysis performed is based upon each specific radioisotope and the exposure potential. Because of the small quantities of radioactive material in most samples, qualitative analyses are not feasible, and only quantitative analyses for predetermined isotopes are performed routinely.

In most cases bio-assay data require interpretation to determine the dose to the person; computer programs are used for evaluation of extensive data on urinary excretion of ^{239}Pu. An estimate of dose is made for all cases in which it appears that one-third of a body burden, averaged over a calendar year, may be exceeded.

6.3.2 Whole Body Counter - The whole body counter (an in vivo gamma spectrometer) may be used for determining internally deposited quantities of most of the gamma ray-emitting substances, and many of the more energetic beta-emitting substances. Thus, it provides a direct method of determining body burdens of those substances.

6.4 Records and Reports

Most records and reports are prepared by electro-data processing (EDP) techniques through the use of high-speed digital computer systems. The IBM 7090, located at the Central Data Processing Facility (CDPF), turns out routine weekly, quarterly, and annual reports involving external dose data. (A typical weekly report is shown in Figure 29; a typical quarterly report is shown in Figure 30.) An IBM 1401, located at the Y-12 Plant, is used to provide the weekly pocket meter report (see Figure 31). Quarterly bio-assay listings are prepared by the IBM 7090 at CDPF; a weekly Bio-Assay Sample Status Report (Figure 32) is processed by the ORNL Math Panel utilizing a CDC 1604.

A monthly report based on preliminary results of analysis by the whole body counter (IVGS) is prepared by the IBM 7090 at CDPF.

Reports involving AEC-ORO and TVA-EGCR film services are generated by the IBM 7090 at CDPF. Data generated from films used in the temporary tag-pass meter are used in an annual report prepared by use of the IBM 1401 located at the Y-12 Plant.

Body burden estimates of ^{239}Pu are prepared in report form (usually quarterly) by use of the IBM 7090 at CDPF.

Permanent files are maintained at Applied Health Physics Headquarters for each individual who is assigned an ORNL badge-meter. An IBM card cross-indexing system is maintained at the principal monitoring stations for the purpose of expediting meter assignments. These IBM cards are compatible with the various computer programs and provide for the internal audit of all personnel monitoring record data.

Name	ID Number	Symbol	Dosimetry Dates Wk-Yr	Dosimetry Dates Qtr-Yr	Meter Dose DS	Meter Dose DC
Last Name, Initials	PR. No.	PF	35-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	31-63	3-63	0.120	0.090
Last Name, Initials	PR. No.	PF	30-63	3-63	0.030	0.000
Last Name, Initials	PR. No.	PF	36-63	3-63	0.070	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	36-63	3-36	0.370	0.310
Last Name, Initials	PR. No.	PF	32-63	3-63	0.000	0.000
Last Name, Initials	PR. No.	PF	33-63	3-63	0.040	0.020
Last Name, Initials	PR. No.	PF	34-63	3-63	0.260	0.130
Last Name, Initials	PR. No.	PF	35-63	3-63	0.040	0.010

Fig. 29 Typical ORNL Film Monitoring Data.

HEALTH PHYSICS DIVISION
DEPARTMENT 3193
RADIATION SURVEY

Name	ID Number	Symbol	Date Wk-Yr	REM DS	REM DC	REM This Qtr DS	REM This Qtr DC	REM This Yr DS	REM This Yr DC	Total REM DC	A	DC/A
----	-----	PF PN	39-63 39-63	0.860 0.000	0.630 0.000	0.860	0.630	1.68	1.32	35.59	18	2.02
----	-----	PF	39-63	0.340	0.240	0.340	0.240	0.34	0.24	0.24	1	0.80
----	-----	PF	39-63	0.020	0.010	0.020	0.010	0.02	0.01	5.21	14	0.38
----	-----	PF	39-63	0.070	0.040	0.070	0.040	0.30	0.19	18.38	16	1.19
----	-----	PF	39-63	0.390	0.310	0.390	0.310	1.40	1.14	2.74	20	0.14
----	-----	PF	39-63	0.350	0.150	0.350	0.150	0.77	0.49	9.60	17	0.56
----	-----	PEL PF PN	27-63 39-63 39-63	0.010 0.140 0.000	0.010 0.110 0.000	0.150	0.120	0.27	0.24	5.55	6	1.09
----	-----	PF	39-63	0.400	0.200	0.400	0.200	0.73	0.45	7.43	12	0.64
----	-----	PF	39-63	0.180	0.150	0.180	0.150	0.60	0.49	8.43	7	1.34
----	-----	PF PN	39-63 39-63	0.330 0.030	0.110 0.030	0.360	0.140	0.81	0.34	3.00	13	0.24
----	-----	PF PN	39-63 39-63	0.180 0.000	0.080 0.000	0.180	0.080	0.51	0.33	29.82	18	1.68
----	-----	PF PN	39-63 39-63	0.320 0.000	0.270 0.000	0.320	0.270	1.14	0.98	22.76	13	1.76
----	-----	PF	39-63	0.420	0.290	0.420	0.290	1.85	1.11	15.86	16	1.04
----	-----	PF	39-63	0.320	0.140	0.320	0.140	0.67	0.46	8.96	11	0.84
----	-----	PF	39-63	0.390	0.210	0.390	0.210	1.21	0.72	33.62	18	1.87

Fig. 30 Typical ORNL Personnel Radiation Exposure Record.

Dept. 3193 HP Wk. 39

Name	Pr. No.	DC	M	T	W	T	F	S	S	Wk.	Qtr.	Symbol
----	-----		0	0	20	10	0			30	295	D
----	-----		10	5	5	0	0			20	350	
----	-----			0	0	0	0				155	
----	-----		0	0	0						65	
----	-----		15	10	10	10	10			55	385	
----	-----		0	0	0	0	0				70	
----	-----		10	0	10	10	0			30	175	
----	-----		0	0			0	0			185	
----	-----		10	0	0	0				10	40	
----	-----											
----	-----		0	0							40	
----	-----		10	0	5	0	0			15	220	
----	-----		10	0	10	0	0			20	225	
----	-----		10	5	0	0	0			15	110	
----	-----				0	0	0	0			115	
----	-----		40	90	0	0	0			130	260	DW
----	-----		0	10	0	10	0			20	270	
----	-----		15		0	0	0			15	225	
----	-----		10	5	0	10	0			25	275	
----	-----				0	0					140	
----	-----		10	10			0	0		20	250	
----	-----		0	0	0	0	5			5	170	
----	-----			5	0	0	0			5	770	
----	-----		0	0	0	20				20	135	D
----	-----		0		0	0	0				145	
----	-----		5	5		0	0			10	395	
----	-----		0	10	10	0	10			30	260	
----	-----		0	10	10	10				30	400	
----	-----										35	
----	-----		0	5	0	0	0			5	140	

Fig. 31 Typical Pocket Meter Weekly Report.

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Div. Code	Name	Payroll Number	HP Area Number	Type Analysis	Receipt Date	Type Sample	Sample Priority	d/m/Sample	d/m/24 hrs.
HP	----	-----	3550	GUO	12-16-63	U	3		0
HP	----	-----	3019	GUO	12-12-63	U	3		0
HP	----	-----	3019	GUO	12-16-63	U	3		0
HP	----	-----	3019	GUO	12-12-63	U	3		0
Div. Total 4									

Fig. 32 Typical Weekly Bio-Assay Sample Status Report.

Copies of the EDP reports, both temporary and final, are maintained for both the internal and external dose programs. Data used in the EDP program are stored on computer quality magnetic tapes. Data pertinent to the work of the dosimetry groups and information used in the non-EDP reports are maintained in record form by Dose Data personnel.

6.5 Program Developments

During 1964 the computer program for pocket meter readings was modified slightly to provide (in addition to the weekly departmental report) a quarterly summary report and a weekly and quarterly report by employee name, without regard to department.

A quarterly summary of the results of analysis for bio-assay samples was programmed for the computer during the year. An annual summary may be prepared using the same EDP program.

A computer program which details and summarizes the preliminary results of analysis of employees by the whole body counter was prepared during 1964. Reports can be made monthly, quarterly, or annually.

Several changes were made in the badge-metering programs during 1964. Most of these changes resulted in reduced costs, improved appearance of the badge-meter, and increased dosimetric integrity. Among the changes were:

1. A badge-metering system was initiated for use in the Biology Division's facilities at Y-12. The single badge-meter replaced a laminated pass and a temporary tag meter, which had been used for visitors to those facilities.

2. A badge-metering system was initiated for use by persons (a) who make deliveries to Laboratory facilities or (b) who perform non-radiation-associated services within ORNL facilities. These badges are issued at vehicular entry points and are reissued during periods of one month. They replace single-use temporary tag meters.

3. A badge-meter service was initiated to replace the temporary tag passes used by AEC employees on visits to ORNL.

4. The use of manila tag meters for ORNL visitors badged at the Y-12 Plant was discontinued. A badge-meter for such visitors is issued as required, by ORNL Health Physics at Y-12.

Changes 2., 3., and 4 above have led to a reduction in film usage from about 6000 films per year to about 1000 films per year.

5. Consultants to ORNL who are assigned photo-badges and persons whose term of employment or assignment does not exceed three months are being assigned only one badge-meter, rather than two as are assigned to regular employees and long-term assignees for exchange efficiency. This is a saving of more than ten dollars per person badged.

6. A badge-metering system for accident-type monitoring was initiated for persons who are members of guided tours. These persons had not been monitored heretofore except that the person who guided the tour was badged.

7.0 LABORATORY OPERATIONS MONITORING

Radiation incidents are classified according to a severity index system developed over the past five years.[7] The method serves to index unusual occurrences according to degree and permits a ready analysis regarding health physics practices among Laboratory operations. This report summarizes the unusual occurrence frequency rate and discusses some of the problems encountered among Laboratory facilities.

7.1 Unusual Occurrences

There has been a steady decrease in the frequency of unusual occurrences since the classification system was begun in 1960 when 87 such occurrences were recorded. During 1964 the frequency dropped to a five-year low of 29 events (see Table 13) and only about 50 percent of these events were classified as significant. For the purposes of this report, an event is designated as significant when it is such as to (1) exceed a recommended maximum permissible limit and/or (2) require a work stoppage in an operation while clean-up measures are instituted following a radioactive contaminant release. Although about half of the 29 events were classified as significant according to the above definition, no event occurring during 1964 required more than minor inter-departmental assistance before normal operations were resumed. There were no reportable over exposures of personnel although some minor work restrictions were imposed. The frequency rate within the Laboratory divisions showed a corresponding drop during 1964 over the previous four years (see Table 14). The same was true within the particular operating facilities (see Table 15) and the variations noted are known to depend upon the quantity of radioactive materials handled, the number of radiation workers involved, and the radiation hazard potential associated with a particular operation or facility.

7.2 Radiation Surveys

During 1964, the Radiation Survey personnel assisted the operating groups in keeping the contamination, air concentration, and personnel exposure levels well below the maximum permissible established limits. Through discussion, seminars, safety meetings, and informal discussions with supervision of the operating groups, they assisted in reducing or eliminating a number of problems associated with radiation protection at the Laboratory. The following is a brief description of a few of the problems and methods of solution.

7.2.1 Disposal of Radioactive Waste from the Fission Products Pilot Plant - The transfer of solid radioactive waste from the Fission Products Pilot Plant to the burial ground area has presented a problem since the facility started processing kilo-curie quantities of fission product

[7] See Applied Health Physics Annual Report for 1963, ORNL 3665, pp. 14-15.

Table 13 UNUSUAL OCCURRENCES SUMMARIZED FOR THE 5-YEAR PERIOD ENDING WITH 1964

	1960	1961	1962	1963	1964
Number of Unusual Occurrences Recorded	87	75	55	43	29
A. Number of incidents of minor consequence involving personnel exposure below MPE limits and requiring little or no clean-up effort	27	34	25	11	14
B. Number of incidents involving personnel exposure above MPE limits and/or resulting in special cleanup effort as the result of contamination	60	41	30	32	15
1. Personnel Exposures	10	7	7	4	9
a. Nonreportable overexposures with minor work restrictions imposed	9	6	7	3	9
b. Reportable overexposures with work restrictions imposed	1	1	0	1	0
2. Contamination of Work Area	59	40	30	32	15
a. Contamination that could be handled by the regular work staff with no appreciable departmental program loss	56	37	28	30	14
b. Required interdepartmental assistance with minor departmental program loss	2	3	2	2	1
c. Resulted in halting or temporarily deterring parts of the Laboratory program	1	0	0	0	0

Table 14 UNUSUAL OCCURRENCE FREQUENCY RATE WITHIN THE DIVISIONS FOR THE 5-YEAR PERIOD ENDING WITH 1964

Division	No. of Unusual Occurrences 1960	1961	1962	1963	1964	5-Year Total	Per Cent Lab. Total (5-Year Period)
Analytical Chemistry	4	3	5	9	3	24	8.3
Biology	2	1	1	2		6	2.1
Chemical Technology	17	19	13	11	3	63	21.8
Chemistry	3	2				5	1.7
Plant and Equipment	5	4	3	1	2	15	5.2
Inspection Engineering			1		1	2	0.7
Electronuclear Research	5	7				12	4.1
Health Physics	1			1*	1	3	1.0
Instrumentation and Controls	1				1	2	0.7
Isotopes	14	9	18	5	12	58	20.1
Metals and Ceramics	3	5	2	1		11	3.8
Neutron Physics	2	3	3	2		10	3.4
Operations	14	12	6	9*	3	44	15.2
Physics		1	2	3	3	9	3.1
Reactor	11	7				18	6.2
Reactor Chemistry	1	1				2	0.7
Solid State	3		1			4	1.3
Thermonuclear		1				1	0.3
Construction	1					1	0.3
Totals	87	75	55	43	29	289	100.0

*Shared responsibility with another division for one unusual occurrence.

Table 15 UNUSUAL OCCURRENCES CLASSIFIED ACCORDING TO THE OPERATING FACILITY IN WHICH THEY OCCUR FOR THE 5-YEAR PERIOD ENDING WITH 1964

Building or Facility	Number Recorded					5-Year Total
	1960	1961	1962	1963	1964	
2000	2	1	1			4
2001	1					1
2005		1				1
2007	1					1
2024		1				1
2523				1		1
2528	1					1
3001	3	2	2			7
3005	4	1		2	1	8
3010			1	1		2
3012	1					1
3019	11	16	9	17	5	58
3025	3	2	2	2	1	10
3026-C	2	1				3
3026-D		2				2
3028	2	2	2		1	7
3029	2	1	1			4
3031		1			1	2
3032			2			2
3033	1		2			3
3038			3	1	2	6
3042	3	5	3	3		14
3044	1					1
3047				1		1
3074					1	1
3077					1	1
3500	1					1
3504					1	1
3505		1				1
3508	2	1	1		1	5
3517	6	3	8	3	5	25
3550	1		2	2		5
4500N	2	2	1	1		6
4500S				1		1
4501	3	5				8
4507		2	3	1		6
4508				1		1
5500			2	1	3	6
7500	5	7				12
7503					1	1
7700			2	1		3
9201-2	4	8			1	13
9204-1	6	1			1	8
9204-3	4		1		2	7
9207	2	1	1	2		6
9213	2	2	1			5
9733-3	1					1
9766			1			1
Misc.	10	6	4			20
Burial Gn. # 5				1	1	2
S. Tank Farm				1		1
GRAND TOTAL	87	75	55	43	29	289

material. The problem arose primarily because of the difficulty of adequately packaging and containing this highly radioactive particulate material until properly disposed of in the burial ground. An incident at the burial ground early in 1964 which resulted in some personnel exposure stimulated an extended effort to control the problem. A procedure was instituted to decontaminate the inside as well as the outside of the specially constructed, lead shielded, Dempster Dumpster pan prior to its use. This procedure prevents accumulation of large amounts of radioactivity inside the pan which has a tendency to become airborne during the disposal operation. Wastes to be transferred are doubly contained by placing the material in buckets which are then placed in heavy gauge plastic bags which are properly sealed before removal from the cell. A motor crane is now used, instead of a dumpster truck, which allows the pans to be carefully lowered into the burial ground trench. The new procedure minimizes the possibility of airborne contamination due to jarring and other impact conditions. Since the early part of this year when these procedures were placed in effect, no significant spread of contamination has been detected.

7.2.2 Installation of Filters on Vents at Tank Farm - During the past few years, the movement of radioactive liquid waste between tanks in the tank farm area by steam jetting methods of transfer has resulted in the release of various amounts of radioparticulate matter to the local area. During the past year filtered vents were installed on the tanks which greatly reduced the air and ground contamination in the tank farm area. Although the addition of the filtered vents has proven to be a great improvement, further improvements are being considered and preliminary studies are underway to determine the feasibility of connecting all liquid waste tank vents to the main laboratory off-gas system.

7.2.3 Release of Iodine and Tellurium from Short-Cooled Material - During 1964 the Volatility Pilot Plant (VPP) processed spent Low Intensity Testing Reactor (LITR) and Oak Ridge Research Reactor (ORR) fuel elements with cooling periods ranging from one and one-half years to less than four weeks. The processing of such short-cooled material created a monitoring problem and required a more rigid control over the release of fission gases (principally iodine and tellurium). Charcoal traps were installed in the VPP off-gas to aid in the removal of iodine and tellurium. Data obtained from processing charcoal traps installed in the stack monitors indicated that the release of iodine and tellurium were well below established maximum permissible levels.

7.2.4 Pressure Differential Problem in Interim Alpha Laboratory - The construction of the Interim Alpha Facility in the basement of Building 3019 was completed and the facility placed in operation during 1964. The use of a single header to connect glove boxes in this facility with glove boxes located on the first floor resulted in a pressure differential problem. On one occasion the glass front panel on a glove box was shattered due to excessive pressure while adjustments were being made on the glove box off-gas header. The problem was eliminated by the installation of pressure relief valves in the off-gas headers of the first floor laboratories and the Interim Alpha Facility.

7.2.5 Control of Personnel Entry to Multicurie Gamma Source Rooms and X-Ray Machines - During the year a procedure was initiated in the Biology Division to be followed by all personnel having valid reasons for using multicurie gamma sources and X-Ray machines. A thorough study of source facilities and operating conditions had indicated the desirability of a detailed training program to assure the maximum protection of personnel. All personnel having Division approval to use large sources of radiation must progress through a training period until the principal scientist in charge of the source facility is satisfied that the person is capable of operating the facility without direct supervision. A primary requirement is that all source room facilities are kept locked when not in use.

7.2.6 Renovation of Bulk Shielding Reactor Pool - In July, 1964, the Bulk Shielding Reactor (BSR) pool was drained in order to repair and renovate the tank. This involved the stopping of leaks and the repainting or resealing of the inner surface of the tank. Radiation dose rates (to 600 mR @ 3") and radiocontaminants (primarily ^{90}Sr and ^{89}Sr, to 10^5 d/m per 100 cm^2) were reduced to permissible working levels by sandblasting techniques. Workmen experienced no contamination or radiation exposure which exceeded the acceptable limits.

7.2.7 Release of Air Activity at Low Intensity Testing Reactor - During 1964 there were intermittent periods of air activity detected in the LITR. This was especially true in the west room where the activity appeared to come in bursts of short duration but high in concentration. Analysis showed that the air activity was principally ^{138}Cs with some ^{88}Rb, and that the prime source was reactor water leaking into the sub-pile room. The "P" tube opening in the west room was determined as the access area of the fission gasses into the room. The recommendation was made that the reactor water leaks into the sub-pile room be stopped, the negative pressure in the sub-pile room be increased, and that the "P" tube opening in the west room be sealed and off-gassed. Attempts made to stop the water leaks were only partially successful. The negative pressure in the sub-pile room stayed essentially the same up until the time it was connected into the new building vent system when it increased from approximately .1" H_2O to .4" H_2O. The "P" tube area in the west room was sealed and off-gassed. This alleviated the problem. When the building vent system was installed at the LITR, the sub-pile room was kept at a greater negative pressure than the west and the east rooms. This allowed removal of the temporary off-gas line at the west room contained area. Air activity levels are now normal in both the west and east rooms.

7.2.8 Release of Air Activity at Oak Ridge Research Reactor - In March of 1964, the ORR was experiencing air activity levels in the building up to a factor of 10 greater than normal when the reactor was being brought up to power following a short shutdown to refuel. The activity was principally ^{88}Rb and the access areas into the building were found to be the pool overflow drain holes. The overflow holes were sealed as a temporary measure. Later, a permanent device was installed which allows the gas to be vented under the pool water.

7.2.9 Tritium Contamination at High Voltage Accelerator Laboratory - During a greater part of 1964 the target room surfaces in Building 5500, where ^3H is handled, remained chronically contaminated to levels which substantially exceeded the established operating limits of 2000 ^3H d/m/100 cm^2. The following factors contributed to these conditions: (1) it is very difficult to design gaseous tritium targets which have sound containment features; (2) foil type targets (Ti^3H and Zr^3H) must be handled with extreme caution while at target position in order to prevent gross contamination due to "flaking" of targets; (3) target positions are not provided with adequate out-gassing facilities; (4) dismantled tritium contaminated beam piping contributes to the contamination problem in storage areas (usually target room).

To aid in controlling these conditions some steps have been taken, and recommendations made as follows: (1) target areas, while ^3H targets are in use, are designated as Contamination Zones with recommended precautions and safeguards that the job requires; (2) daily urine samples are required of participants actively engaged in target handling operations; (3) the contaminated beam piping be sealed at each end and stored in designated areas. Also, that (4) targets be stored in a hood or controlled ventilation area when not in use. These recommendations have not been fully implemented at the present time.

A new tritium monitoring system for detecting and evaluating the presence of tritium contamination is currently under study.

7.3 Laundry Monitoring

A total of 818,561 articles of wearing apparel were monitored at the laundry during 1964. About 1 percent of the items (8,822) were found to be contaminated. Of the 323,431 khaki garments monitored during the year, only 318 were found contaminated.

8.0 LABORATORY ASSAYS

Laboratory Assays Units provide laboratory support to the Applied Health Physics Monitoring Sections. These services include (1) the analysis of body fluids and excreta (bio-assay) for the monitoring of personnel for internal radiation exposure, (2) the radiochemical analysis of environs samples, (3) counting services for the environs monitoring and radiation survey programs, (4) autoradiography, and (5) whole body counting (in vivo gamma spectrometry).

8.1 Bio-Assay Analysis

The number and types of analyses performed by the Bio-Assay Unit during 1964 are given in Table 16. A total of 7542 analyses were performed which include 7148 analyses on samples submitted by donors and 394 analyses on standard and blank samples analyzed for control purposes. Approximately 91 percent of the samples were analyzed for either gross alpha, strontium, or plutonium. The total number of analyses on samples submitted during 1964 decreased by about 9 percent from the number processed during 1963.

A new ion exchange procedure for the analysis of alpha emitting isotopes (U, Np, Pu and transplutonium elements) developed by the Bio-Assay group was installed in March of 1964 as a replacement for the old gross alpha procedure. This procedure permits multiple analyses on a single sample and resulted in a reduction in the cost of chemicals and reagents used for analysis of alpha emitting isotopes. It is estimated that a savings of approximately $2500 per year may be realized in changing to this technique.

8.2 Counting Facility

A total of 297,080 samples were processed by the counting facility during 1964. A tabulation of the number and types of samples counted is presented in Table 17. This total represents about a 14 percent decrease in the number of samples processed as compared with the previous year.

In the early part of the year, the counting facility's single channel gamma spectrometer was equipped and standardized for the determination of ^{131}I affixed to the ion exchange resin used in the separation of ^{131}I from environs milk samples.

8.3 Environs Monitoring Sample Analysis

Table 18 presents the number and type of environs samples analyzed and the type of analysis performed on each type of sample. A total of 11,946 environs samples were analyzed during 1964 as compared to 13,614 samples analyzed in 1963. The decrease resulted largely from the curtailment of the autoradiographic program for a number of in-building continuous air monitors. Analysis of environs monitoring samples may

Table 16 BIO-ASSAYS ANALYSES - 1964

Analytical Procedure	Number of Analyses	
Urine:		
Gross Alpha	697	
Sr	2,659	
U	1,095	
TRE (total rare earths)	59	
^{3}H	187	
^{137}Cs	98	
^{239}Pu	2,011	
^{106}Ru	4	
^{32}P	15	
Other	90	
Total		6,915
Fecal:		
Gross Alpha	100	
Sr	115	
TRE (total rare earths)	4	
U	13	
Total		232
Miscellaneous:		
Blood, sputum, breath		1
Standards and blanks		394
GRAND TOTAL		7,542

Table 17 COUNTING FACILITY RESUME - 1964

Types of Samples	Number of Samples			Unit Total	Weekly Average
	Alpha	Beta	Gamma		
Survey Area Samples					
Smears	119,789	125,869		245,658	4,724
Air Filters	21,277	21,277		42,554	818
Environs Monitoring					
Air Filters	2,184	2,184		4,368	84
Gummed Paper		1,165		1,165	22
Rain Water		1,164		1,164	22
White Oak Lake Effluent	218	1,068		1,286	25
Animal Thyroids			214	214	4
Milk			251	251	5
Miscellaneous	210	210		420	8
GRAND TOTAL	143,678	152,937	465	297,080	5,712

Table 18 ENVIRONMENTAL MONITORING SAMPLES - 1964

	Sample Type	Type of Analysis	Number of Samples
1.	Monitoring network filters	Gross beta, autoradiogram	2,530
2.	Gummed paper fall-out trays	Gross beta, autoradiogram	1,307
3.	CAM filters	Gross beta, autoradiogram	5,567
4.	Rain water	Gross beta	582
5.	White Oak Dam effluent	Gross beta, radiochemical, gamma spectrometry	1,015
6.	Clinch River water	Gross beta, radiochemical gamma spectrometry	20
7.	Raw milk	Radiochemical	410
8.	Pasture grass	Radiochemical, gamma spectrometry	200
9.	Potable water	Radiochemical, gamma spectrometry	56
10.	Silt composites	Radiochemical, gamma spectrometry	45
11.	Animal thyroids	Gamma spectrometry	214
		TOTAL	11,946

range from a single determination to as many as ten determinations per sample depending upon the radionuclides present. The methods used by the various analytical groups are generally described in the ORNL Master Analytical Manual.

The method for processing milk samples was changed from a precipitation type technique to ion exchange separation during the early part of the year. This change in technique increased the reliability of the analysis results and reduced the processing time, thus permitting the handling of additional samples.

8.4 Autoradiography

A total of 2,188 films were processed during 1964 in support of radioparticulate studies conducted by the Environs Monitoring Units.[8]

8.5 Whole Body Counter[9]

During the calendar year 1964 the whole body counting program included 1644 human counts on 1483 persons; 1342 of the counts, or approximately 82 percent, showed a normal human spectrum. The number of counts made this year represents an increase of approximately 56 percent over the number of counts made in 1963. In addition, from one to two days a week have been reserved for special counts, calibrations, research and development and routine maintenance. Table 19 lists the number of cases and quantity of isotopes detected in the counts which showed activity other than normal ^{40}K, and higher than the average normal ^{137}Cs background (.025 μc ^{137}Cs for 1964).

The whole body counting program this year has continued the idea of having at least an initial or baseline count on essentially every person with a potential for future exposure. This has been implemented by having the Radiation Control Officer in each division select the persons to be counted and schedule counting time. In each case, the primary emphasis has been placed on counting first those with the greatest potential for exposure. At times the routine schedule has been interrupted or changed where necessary to permit counting persons suspected of having sustained an actual exposure to radioactive materials. In these cases schedule changes have been accomplished smoothly by telephone contact with the Radiation Control Officer of the division being counted. The program of obtaining baseline counts was essentially completed, and future scheduling will place greater emphasis upon the current exposure potential of each individual.

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[8] Methods described in ORNL-2601, "Radioactive Waste Management at Oak Ridge National Laboratory".

[9] The Whole Body Counter is operated by the Health Physics Technology Section.

Table 19 MEASURABLE RADIOACTIVITY FOUND IN ROUTINE WHOLE BODY MONITORING PROGRAM - CALENDAR YEAR 1964

Isotope	Maximum Amount Detected (μc)	Percent MPBB	Number of Counts
^{51}Cr	0.006	< 0.001	10
^{58}Co	0.010	< 0.05	32
^{60}Co	0.015	< 0.1	20
^{65}Zn	Trace		1
^{75}Se	0.029	< 0.1	11
^{90}Sr-^{90}Y	1.35	88.8	45
^{95}Zr-^{95}Nb	0.044	0.22	56
^{106}Ru-^{106}Rh	0.026	1.3	36
^{125}Sb	0.646	1.6	28
^{131}I	0.022	3.1	11
^{137}Cs	0.135	0.67	223*
^{144}Ce-^{144}Pr	Trace		2
^{235}U	Trace		1
Indications of Bremsstrahlung			26
Unidentified			8

*During this year $\leq$ 0.025 μc ^{137}Cs was considered "normal" or likely from dietary intake only.

During this report period, the whole body counter was used to estimate the ^{90}Sr-^{90}Y body burden of several ORNL employees who had been involved in airborne particulate contamination incidents. In two of the incidents, the contamination was $^{90}SrTiO_3$, in another the contaminant was most likely $^{90}SrCO_3$. In cases of known exposure, the bremsstrahlung continuum was readily discernible at levels of around 5 percent MPBB and higher, although at these relatively low levels, whole body count estimates of ^{90}Sr body burden are subject to large uncertainties. It was found that scan count estimates of body burden are quite sensitive to differences in distribution of the deposit within the body relative to the body surface, hence any disparity between the distribution of the actual deposit and the standard or calibration deposit introduces errors in estimates. After the majority of the initial GI tract deposit was eliminated by the persons exposed, chest counts taken with the lead collimated 8" x 4" NaI crystal provided the best estimate of ^{90}Sr burden. Figure 33 shows the estimates of chest burden for the only two cases which were calculated to have averaged > 30 percent of the maximum permissible level for the year.

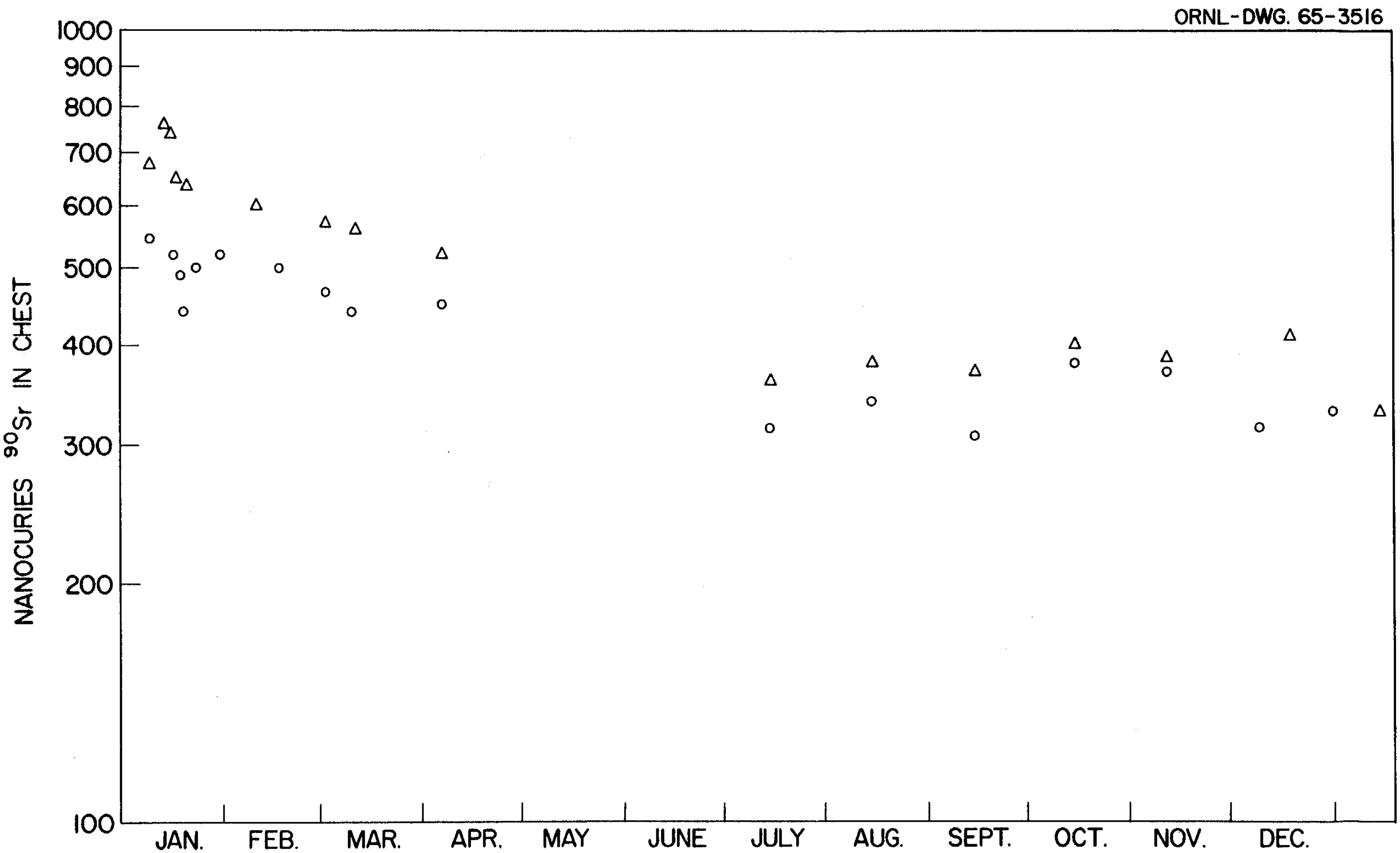

Fig. 33 Chest Burden as Estimated from In Vivo Counting Two Employees Exposed to Airborne Particulate $^{90}SrTiO_3$.

9.0 HEALTH PHYSICS INSTRUMENTATION

The Health Physics Division shares with the Instrumentation and Controls Division the responsibility for the development of electronic radiation monitoring instruments used in the Laboratory health physics program. Normally the Health Physics Division is responsible for determining the need for new instrument types and modifications to existing types, specifies the health physics requirements for design criteria, originates work orders, tests and evaluates the preliminary design, and approves the final design. The Health Physics Division is also responsible for calibrating all instruments used in the health physics program and is allocated the funds for maintenance of these instruments. Maintenance is performed or crossordered by the Instrumentation and Controls Division.

Non-electronic personnel monitoring devices are designed, tested, calibrated, and maintained by Health Physics Division personnel.

9.1 Instrument Inventory

The electronic instruments used in the health physics program are divided, for convenience in servicing and calibrating, into two classes: the first class includes battery-powered portable instruments; the second class includes the stationary instruments that are a.c. powered. Portable instruments are assigned and issued to the Radiation Survey Units. Stationary instruments are the property of the Laboratory division which has the monitoring responsibility in the area in which the instrument is located. Table 20 lists portable instruments on hand at the close of FY 1964; Table 21 lists stationary instruments on hand at the end of 1964. There were net increases in 1964 of 109 portable instruments and 58 stationary instruments.

During 1964, 848 new pocket meters and 373 new fiber dosimeters (200 mR range) were issued by ORNL Stores. Most of the meters issued were replacements for instruments which had been lost or damaged.

Inventory and Service Summaries for health physics instruments are prepared on an IBM 7090. These computer programmed reports enable the Instruments Group to maintain a current inventory on most health physics instrument requirements.

The allocation of stationary health physics monitoring instruments by divisions is shown in Table 22. The divisions are identified by accounting codes. Discrepancies between Table 22 and Table 21 are mainly because of differences in inclusion of "other" or miscellaneous instrument types and to what extent instruments in ORNL facilities at the Y-12 Plant are included.

Table 20 PORTABLE INSTRUMENT INVENTORY - 1964

Instrument Type	Working Inventory July 1963	Instruments Acquired FY 1964	Instruments Retired FY 1964	Working Inventory July 1964
GM Survey Meter	341	44	1	384
Cutie Pie	349	37	0	386
Juno	36	1	0	37
Alpha Survey Meter	125	28	0	153
Neutron Survey Meter	40	2	0	42
Miscellaneous	17	0	2	15
TOTAL INVENTORY	908	112	3	1017

Table 21 INVENTORY OF STATIONARY, RADIATION MONITORING INSTRUMENTS FOR THE YEAR 1964

Instrument Type	Installed During 1964	Retired During 1964	Total Dec. 31, 1964
Air Monitor, Alpha	15	0	63
Air Monitor, Beta	18	6	160
Air Monitor, Environmental	1	0	37
Hand-Foot Monitor	2	0	24
Lab Monitor, Alpha	10	3	89
Lab Monitor, Beta	20	1	143
Monitron	10	4	210
Other	1	5	22
TOTAL	77	19	746

Table 22 HEALTH PHYSICS FACILITY MONITORING INSTRUMENTS DIVISIONAL ALLOCATION - 1964

ORNL Division	α Air Monitor	β Air Monitor	α Lab Monitor	β Lab Monitor	Monitron	Other	Total
Analytical Chemistry	3	12	8	15	14	6	58
Chemical Technology	23	34	27	10	29	24	147
Chemistry	2	6	9	12	19	6	54
Metals and Ceramics	6	15	13	17	11	8	70
Reactor	6	12		9	11	5	43
Isotopes	13	33	6	31	52	17	152
Operations		35		11	53	13	112
All Others	11	17	17	29	25	59	158
TOTAL	64	164	80	134	214	138	794

9.2 Calibrations Facility

The Health Physics Division maintains a calibration facility for the calibration and maintenance of portable radiation instruments and personnel metering devices. The facility is equipped with calibration sources, remote control devices, and shop space for the use of Instrumentation and Controls Division maintenance personnel. Health Physics personnel assign, arrange for maintenance of, calibrate, provide delivery services for, and maintain inventory and servicing data of all portable health physics survey instruments.

Portable instruments are serviced (1) whenever repairs are needed, (2) at least once each two months for those which have replacement-type batteries, and (3) at least once each three months for those instruments which have "permanent" (rechargeable) batteries. The calibration services performed during FY 1964 on portable instruments (see Table 23) were about 15 percent greater than were performed during the prior year due to the increase in inventory and emphasis on pocket meter reliability (see ORNL-3665, 6.3.4, p. 23, "Applied Health Physics Annual Report for 1963".

Stationary instruments are calibrated by Calibrations Group personnel or by Radiation Survey personnel who use sources which are designed, standardized, and provided by the Calibrations Group.

9.3 Instrumentation Development

During 1964 new designs were initiated and/or completed and/or tested as follows:

1. Large-Area Alpha Scintillation Detector - The large-area alpha scintillation detector (6 in. x 12 in.) previously reported (ORNL-3665, p. 23, "Applied Health Physics Annual Report for 1963") was received and tested. Good sensitivity to alpha particles and very low background was obtained. The detector uses a single two-inch photo-multiplier.

2. Neutron Monitor - The Neutron Monitor and Burst Detector, ORNL Q-2296, was calibrated and subjected to a number of exposures both burst and steady-state operation, at various neutron levels, at the Health Physics Research Reactor.

3. Portable Counter for Alpha - The design and prototype fabrication of a combination count rate and integrating portable electronic package for use with the Q-2101 Alpha Scintillation probe was initiated. This device will permit accurate integration at low counting rates as well as three rate meter ranges for use at intermediate and high counting rates.

4. Alpha Air Monitor - Design and fabrication of a prototype alpha/beta ratio-type, alpha air monitor was initiated and the work was 90 percent completed at the close of the year. This device will permit detection of low levels of airborne, particulate-associated, alpha particle emitters in the presence of normally occurring radon-thoron daughters and moderate levels of gamma radiation.

Table 23 CALIBRATIONS RESUME - 1964

A. Portable Instruments Calibrated

1.	Beta-Gamma	3,923
2.	Neutron	115
3.	Alpha	943
4.	Pocket Chambers and Dosimeters	3,105

B. Films Calibrated

1.	Beta-Gamma	2,272
2.	Neutron	20

5. Pocket Meter Reader - The pocket meter charger-reader devices (Victoreen Minometer II) have an upper reading limit of 200 mR, which is about the upper limit of reading for accuracy within the range $\pm$ 95 percent, because the voltage on the condenser-type pocket chamber becomes rather low at 200 mR. Because a few readings greater than 200 mR were being obtained each week, which usually necessitates processing the user's monitoring film, and because in almost every case it was estimated that the off-scale (> 200 mR) reading was not much in excess of 200 mR, it was very desirable to know if the off-scale reading exceeded only slightly the 200 mR level. Tests were conducted and it was determined that the minometer could be modified so that in addition to the accurate 0 - 200 mR range, a second range could be provided which would reliably indicate if a reading greater than 200 mR was less than 300 mR. This change was incorporated in the instruments and has resulted in a 90 percent reduction in the processing of films because of off-scale pocket meter readings.

6. Portable Counter for Alpha, Modification - The circuitry of the Q-1975, Portable Alpha Counter, was modified to preclude register jamming at high counting rates (> 15 c/s). Prior to the change, the register would cease to operate at high rates, and this might be interpreted as a safe condition. With the modification, the register will operate at the maximum rate of about 15 c/s if the input counting rate exceeds that level. All Q-1975 instruments now have this modification.

7. Portable Gas Proportional Counter - One of the objectionable features of the Eberline PAC-3G, Portable Gas Proportional Counter is that the shut-off valve of the gas supply will not remain in the "off" position because of the pressure in the gas cylinder. A latching mechanism was designed and installed on a number of our PAC-3G's.

10.0 PUBLICATIONS AND PAPERS

H. H. Abee and D. M. Davis, "Radiation Dose Received by Passengers and Crew on Planes Carrying Radioisotope Shipments", presented at International Symposium on the Dosimetry of Irradiations from External Sources, Paris, France, November, 1964.

H. H. Abee, "Environmental Surveys Following Accidental Releases", Nuclear Safety, Vol. 6, No. 1, pp. 87 - 89, Fall 1964.

D. M. Davis, "Action Levels for Radiation Control at ORNL", presented at Annual Meeting of the American Industrial Hygiene Association, Philadelphia, Pennsylvania, April 29, 1964.

E. D. Gupton and D. M. Davis, "Energy Response of the ORNL Cutie Pie Chamber to Photon Radiation", ORNL-TM-897, June 18, 1964.

E. D. Gupton, W. F. Ohnesorge and H. M. Butler, "Calibration of the ORNL Model Q-2240, Continuous Air Monitor", ORNL-TM-864, May 14, 1964.

E. D. Gupton, "Personnel Monitoring", Nuclear Safety, Vol. 5, No. 2, pp. 192-196, Winter 1963-1964.

E. D. Gupton and S. Fukushi, "Energy, Response, Gamma Discrimination and Stability of the ORNL Model Q-2047, Fast Neutron Survey Meter", Health Physics, Vol. 10, pp. 193-194, 1964.

11.0 VISITORS AND TRAINING GROUPS

During 1964, there were 150 visitors to Applied Health Physics, as individuals or in groups, for training purposes. Table 24 is a listing of the training groups which consisted of six or more persons.

Table 24 TRAINING GROUPS IN A.H.P. FACILITIES DURING 1964

Facility	Number	Training Period
ORINS	12	10/22/64 (2 days/wk)
Paducah, Ky. and Y-12	8	
U.N.C. Public Health	6	8/18 to 8/19/64
Southern Universities	20	8/17/64
AEC-Fellowship	24	June - August
ORINS (Public Health)	12	10/22 to 11/19/64
ORINS	9	7/15/64 (tour)
Taft Rad. Health Center	23	3/16/64

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68. W. R. Casto
69. R. L. Clark
70. A. J. Cook
71. J. A. Cox
72. F. L. Culler
73-80. D. M. Davis
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82. L. G. Farrar
83. D. E. Ferguson
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85. J. L. Fowler
86. J. H Frye, Jr.
87. C. B. Fulmer
88. D. C. Gary
89. J. H. Gillette
90. W. Y. Gissel
91. W. R. Grimes
92. E. D. Gupton
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94. J. C. Hart
95. A. Hollaender
96. A. S. Householder
97. J. T. Howe
98. H. C. Hoy
99. T. W. Hungerford
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108. W. H. Jordan
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